COMBUSTION ENGINEERING, INC.

WINDSOR NUCLEAR FUEL MANUFACTURING

LICENSE APPLICATION

PARTS I & II

(LICENSE SNM-1067)

JUNE 15, 1993

COMBUSTION ENGINEERING, INC.
NUCLEAR POWER BUSINESSES
1000 Prospect Hill Road
Windsor, Connecticut 06095

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PART I. LICENSE CONDITIONS

1.0 Standard Conditions and Special Authorizations

1.1 Name

Combustion Engineering, Inc., is incorporated in the State of Delaware with its corporate offices at 900 Long Ridge Road in Stamford, CT. The location where licensed activities will be conducted is at 1000 Prospect Hill Road in Windsor, CT.

1.2 Location

The mailing address for all license correspondence is:

Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, CT 06095

Licensed activities shall be conducted primarily at the Nuclear Fuel Manufacturing facility (Building #17) and an adjacent warehouse and shipping dock (Building #21).

1.3 License Number

Activities are covered by the License SNM-1067; Docket 70-1100.

1.4 Possession Limits & Location

Combustion Engineering, Inc., requests authorization to receive, use, possess, store and transfer at its Windsor site, the following activities of radioactive materials.

	Isotope	Form	Quantity	Location
1)	Uranium enriched to ≤5.0% weight percent U235	Uranium Oxides	500,000 Kg U	Manufacturing-Bldgs. #17 & #21 & storage in trailers adjacent to Bldgs #17 & #21.
2)	Uranium enriched to less than 20 weight percent U235	Any	4800 gms U^{235}	Bldg. 17, & 21 (Bldg. 17 & 21 limited to 350 gm U235 each for enrichments exceeding 5.0 weight percent U235).
3)	Natural and/or Depleted Uranium	Any	10,000 KgU	Bldg. 17 & 21
4)	Pu238	Encapsulated Neutron Sources	5 sources, each containing less than 2.0 gm Pu238	Building #17
5)	Pu	Any Form	160 micrograms as analytical samples	Bldg. 17, & 21
6)	Encapsulated Neutron sources	U_3O_8	20 sources, each containing 1.7 gm U235	Bldg. 17 & 21
7)	Uranium enriched to or greater than 20 weight percent U235	Residue	1000 gms U235	Windsor Site

1.5 Section Deleted

1.6 Authorized Activities

The primary activities carried out with special nuclear material in buildings at the Windsor site include, but are not limited to the following:

Bldg. 17 - Manufacture of fuel assemblies utilizing low enriched uranium (up to 5.0 wt% U-235) in the form of uranium oxide powder, pellets, rods, and in assemblies.

Bldg. 21 - Storage of SNM in shipping containers.

Windsor Site - Residue from prior operations, not to exceed 350 gms U-235 in any one location. Additional locations to be separated from one another by a minimum of 12 feet.

1.7 Exemptions and Special Authorizations

Licensed activities in Bldgs. 1, 1A, 2, 2A, 3, 3A, 5, 6, 16, and 18 shall be of a product development nature and the material may ultimately be returned to the Nuclear Fuel Manufacturing facility. These transfers shall not require the issuance of applicable NRC transfer documents, but shall be transferred in accordance with the provisions of this license, and shall be handled as a departmental transfer and shall be controlled by the Fundamental Nuclear Material Control Plan (FNMC) referenced in Section 9.0 of this application.

In each area where labelling of all containers is not in accordance with 10CFR20.203(f), all personnel ingress locations shall be posted with a sign stating that "Every container or vessel in this area, unless otherwise identified, may contain radioactive material."

2.0 Organization and Administration

The President, Nuclear Fuel has the ultimate responsibility for ensuring that corporate operations related to Nuclear Fuel are conducted safely and in full compliance with applicable Federal, State and local regulations, licenses and certificates of compliance. The President has delegated this safety and compliance responsibility for nuclear fuel manufacturing to the Vice President, Manufacturing Operations, who has in turn delegated this responsibility to the Uranium Plant Manager.

2.1 Organization Responsibilities and Authority for Key Positions Important to Safety

2.1.1 Uranium Plant Manager

The Uranium Plant Manager, Windsor Nuclear Fuel Manufacturing reports to the Vice President, Manufacturing Operations. He or she has the overall responsibility for the safe conduct of all activities that are regulated by the Nuclear Regulatory commission in Combustion Engineering's nuclear fuel manufacturing responsibilities encompass the following functions: operations, accountability, security, training, criticality, radiological and industrial safety, environmental protection, transportation, materials handling and storage, licensing, process and equipment engineering and maintenance. These responsibilities are administered through the professional staff of the Windsor Nuclear Fuel Manufacturing Facility and the technical staff of Combustion Engineering's Nuclear Fuel Department.

2.1.2 (SECTION DELETED)

2.1.3 Senior Criticality Specialist

The Senior Criticality Specialist may be a member of the technical staff of the Nuclear Fuel organization or an outside consultant who reports in a functional manner to the Chairperson of the Facility Review Group. He or she provides assistance to the Group in executing those aspects of the Group's function which relate to criticality safety. The Senior Criticality Specialist also serves as the second reviewer for criticality evaluations performed by the Nuclear Criticality Specialist (first reviewer).

The Senior Criticality Specialist has no production responsibilities. The Senior Criticality Specialist has the authority to halt any operation in the fuel manufacturing facility or product development laboratories that he or she believes to represent an unsafe criticality condition. If an operation is halted for a criticality safety reason(s), it shall not be restarted without the concurrence of the Manager, RP&IS or Uranium Plant Manager, and the concurrence of an individual with the qualifications of a Nuclear Criticality Specialist.

2.1.4 The Manager, Radiological Protection and Industrial Safety

The Manager, Radiological Protection and Industrial Safety reports to the Uranium Plant Manager. The function of the Manager is to provide information, advice, and assistance to operating and engineering management to ensure personnel and environmental protection measures are adequate and to keep records documenting safety related facility operations. He or she has the responsibility for defining and implementing programs and standards related to radiological, criticality and industrial safety, environmental protection and emergency planning for both the fuel manufacturing facility and the product development laboratories. The programs and standards address safety criteria, monitoring, procedures and training materials necessary to ensure the protection of employees, the public and the environment. As part of this responsibility, he or she ensures that ALARA is considered in the implementation process. The Manager reviews and approves safety related operating procedures. These responsibilities are executed for both the fuel manufacturing facility and the product development laboratories by working with cognizant management to ensure safety limits and operating procedures are acceptable.

The Manager, Radiological Protection and Industrial Safety has no production responsibility. If the Manager believes any operation in the fuel manufacturing facility or product development laboratories to be unsafe, he or she has the authority to halt that operation. If an operation is halted for a safety reason(s) (other than nuclear criticality) it shall not be restarted without his or her concurrence or the concurrence of the Uranium Plant Manager.

2.1.5 Nuclear Criticality Specialist

The Nuclear Criticality Specialist is a member of the technical staff of the Nuclear Fuel organization and reports in a functional manner to the Manager, Radiological Protection and Industrial Safety. The Nuclear Criticality Specialist works with the cognizant Line Managers to ensure that nuclear fuel manufacturing facility or product development laboratory operations (processes, procedures and equipment) or changes thereto are acceptable with regard to nuclear criticality safety. He or she executes this responsibility by advising cognizant management regarding criticality safety practices, arranges for analyses or reviews and approves changes to processes, procedures or equipment related to criticality safety.

The Nuclear Criticality Specialist has no production responsibilities. The Nuclear Criticality Specialist has the authority to halt any operation in the fuel manufacturing facility or product development laboratories that he or she believes to represent an unsafe criticality condition.

2.1.6 Nuclear Criticality Analyst

The Nuclear Criticality Analyst performs the detailed numerical criticality calculations as prescribed by the Nuclear Criticality Specialist to verify the acceptability of fuel manufacturing facility or product development laboratory processes or equipment.

2.1.7 Supervisor, Radiological Protection and Industrial Safety

The Supervisor, Radiological Protection and Industrial Safety reports to the Manager, Radiological Protection and Industrial Safety. He or she assists the Manager in carrying out his/her duties and is responsible for surveillance of nuclear fuel manufacturing and product development activities related to radiological, criticality and industrial safety, environmental protection and emergency planning. This surveillance ensures that operations are being conducted in accordance with Federal, State and local regulations, the conditions set down in this application and certificates of compliance, as applicable.

The Supervisor, Radiological Protection and Industrial Safety has no production responsibility. If the Supervisor believes any operation in the fuel manufacturing facility or product development laboratories to be outside specified limits or unsafe, he or she has the authority to halt that operation.

2.1.8 Industrial Safety Specialist

The Industrial Safety Specialist reports to the Manager of Radiological Protection and Industrial Safety. He or she acts as a consultant to the Manager on matters relating to industrial safety and environmental protection at the fuel manufacturing facility and product development laboratories. He or she also advises the Radiation Protection and Industrial Safety Technicians in the proper methods of monitoring industrial safety and environmental protection compliance. The Industrial Safety Specialist has no production responsibility.

2.1.9 Radiological Protection and Industrial Safety Technicians

The Radiological Protection and Industrial Safety Technicians report to the Supervisor, Radiological Protection and Industrial Safety. The Technicians are responsible for the day-to-day monitoring of operations at the fuel manufacturing facility and the product development laboratories. 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. The Radiological Protection and Industrial Safety Technicians have no production responsibilities.

2.1.10 Emergency Director

The Emergency Director coordinates the actions of the emergency response team members (for both on- and off-site support). The Emergency Director shall remain in control of emergency operations until the situation is stabilized or terminated depending on the severity of the incident. The Emergency Director has authority to direct recovery operations for any emergency condition which may arise in the Nuclear Fuel Manufacturing facility or Product Development laboratories. The Uranium Plant Manager may act in the capacity of Emergency Director. The Emergency Director may designate qualified alternates.

2.2 Personnel Education and Experience Requirements for Key Positions Important to Safety

2.2.1 Plant Manager

The minimum qualifications for this position are a bachelor's degree in one of the sciences or engineering, with ten (10) years experience, including at least five (5) years in management positions in the nuclear industry.

2.2.2 (SECTION DELETED)

2.2.3 Senior Criticality Specialist

The minimum qualifications for this position shall be a bachelor's degree in one of the sciences or engineering, with two (2) years experience performing the duties of a Nuclear Criticality Specialist.

2.2.4 Manager, Radiological Protection and Industrial Safety

The minimum qualifications for this position are a bachelor's degree in one of the sciences or engineering, with four (4) years experience in health physics, including two (2) years in operational health physics with uranium bioassay measurement techniques, internal exposure controls and radiation measurement techniques. Alternatively, a Master's degree in science or engineering with three (3) years and one (1) year, respectively, are considered equivalent.

2.2.5 Nuclear Criticality Specialist

The minimum qualifications for this position shall be a bachelor's degree in one of the sciences or engineering, with two (2) years experience performing criticality evaluations.

2.2.6 Nuclear Criticality Analyst

The minimum qualifications for this position shall be a bachelor's degree in one of the sciences or engineering, with one (1) year of experience performing criticality analyses. An analyst without the one year experience may perform analyses under the supervision of a qualified Nuclear Criticality Analyst as a means of obtaining the necessary experience.

2.2.7 Supervisor, Radiological Protection and Industrial Safety

The minimum qualifications for this position are a high school diploma with five (5) years direct experience in at least one of the safety related areas within his or her cognizance. Three (3) of the five (5) years of safety related experience shall have been as a senior radiological protection technician. A senior technician is an individual who has had at least one (1) year experience as a radiological protection technician.

2.2.8 Industrial Safety Specialist

The minimum qualifications for this position shall be an associate's degree in industrial safety, with two (2) years of related experience in industrial safety and/or environmental protection.

2.2.9 Radiological Protection and Industrial Safety Technicians

The minimum qualifications for this position are a high school diploma with one (1) year of experience in at least one of the safety related areas within his or her cognizance. Technicians shall also complete a facility specific training program(s) in safety related areas within their area(s) of cognizance.

2.2.10 Emergency Director

The Emergency Director shall be a member of the Nuclear Fuel management team and shall be familiar with the Nuclear Fuel Manufacturing and Product Development processes and facilities. He or she shall be familiar with the emergency plan and the implementing procedures for the Nuclear Fuel Manufacturing facility and Product Development laboratories.

Alternate Emergency Director designees shall be selected from the Nuclear Fuel Manufacturing or Product Development supervisory levels or above. Alternates shall also be familiar with the emergency plan and the implementing procedures for the Nuclear Fuel Manufacturing facility and Product Development laboratories.

2.3 Facility Review Group

The Nuclear Fuel Manufacturing facility and Product Development laboratory operations are monitored by a Facility Review Group (FRG). The FRG reports to the Uranium Plant Manager and is responsible for oversight of safety related operations.

The FRG is composed of senior personnel from the technical staff of Combustion Engineering's Nuclear organizations that have at least five (5) years experience in the nuclear industry. The overall function of the FRG is to review operations on a regular basis. In order to execute these responsibilities, the Group will meet at least quarterly to review operations and more often if deemed necessary by the Chairperson.

As a minimum, the Group shall review the following areas for practice and trends:

Radiological Safety
Criticality Safety
Environmental Safety
Industrial Safety including Fire and Chemical Safety
Emergency Planning
ALARA
Internal Inspection and Audit Reports

The Chairperson of the FRG shall ensure that audits of the radiological, criticality, industrial and environmental safety programs are performed in accordance with Section 2.7.2. Results of these audits shall be part of the review for practice and trends.

In addition, the FRG will review physical facility changes in the Pellet Shop and changes to operations involving radiation and/or nuclear criticality safety if the Manager, Radiological Protection and Industrial Safety or the Nuclear Criticality Specialist considers the change to be significant. Each review of a significant change by the FRG shall result in a recommendation to the Uranium Plant Manager before implementation of the change.

The Chairperson of the FRG, the Uranium Plant Manager or the Vice President, Manufacturing Operations may request the Group to examine other areas deemed appropriate.

The Group may establish subcommittees and/or use consultants, as necessary, to carry out its various responsibilities. Findings, however, shall be those of the Group and not just that of the subcommittees or consultants.

The Group shall issue a quarterly progress report to the Uranium Plant Manager, Director of Product Development, and the Vice President, Manufacturing Operations.

Findings and recommendations (if any) of the Group shall be reported to the Uranium Plant Manager, and the Director of Product Development with copies to cognizant Line Managers. Records of the Group's findings shall be retained for a period of three (3) years from the date of issue or until closure of issues is obtained, whichever is later.

2.4 Approval Authority for Personnel Selection

The Uranium Plant Manager and each of his direct reports, which are in key positions important to safety, shall be approved by the next two (2) levels of management above the position to be filled. Other staff positions are filled following the normal administrative practices of Combustion Engineering, Inc.

Chairmanship of, and membership on, the FRG shall be by appointment of the Uranium Plant Manager.

2.5 Training

In addition to on-the-job training and training in special operational skills, employees and visitors to the Windsor Nuclear Fuel Manufacturing facility participate in formal (classroom) training programs to ensure a basic understanding of facility operations and safety requirements. The degree of training an individual receives is commensurate with the extent to which he or she will require unescorted access to these facilities or will come into contact with nuclear materials or other hazardous materials or operations that are a part of the manufacturing process. Escorted visitors do not require any training. Guidelines shall be established which denote the various types of training conducted and classifications for employees and visitors with respect to what training/refresher training is required. All safety related training shall be conducted by an individual well versed with the specific training subject matter.

2.5.1 Initial Training

Employees and visitors (as necessary) working in the Windsor Nuclear Fuel Manufacturing facility shall participate in a General Employee Training program. This program shall include information necessary for each individual to understand the nature of the work done at these facilities and to perform his or her duties in a safe manner. As a minimum, the General Employee Training program shall cover the following subject areas; 1) Organization and Administration, 2) Facility Description, 3) Quality Control, 4) Security, 5) Industrial Safety, 6) Radiation Safety, 7) Criticality Safety, and 8) Emergency Preparedness. The General Employee Training program for Nuclear Fuel Manufacturing shall be appropriate to the activities conducted in those respective facilities.

Employees and visitors (as necessary) working in the Nuclear Fuel Manufacturing facility Pellet Shop or whose job involves working with unclad nuclear material shall also participate in a Radiation Worker Training Program. This program shall include information necessary for each individual to understand the nature of the work performed in the work area and to perform his or her duties in a safe manner, especially as relates to the handling of unclad nuclear materials. As a minimum, the Radiation Worker Training program shall provide a higher level of detail concerning radiological and criticality safety than the GET program which is appropriate to facility operations.

Individuals that have had prior radiological protection training, consistent with the activities conducted in Nuclear Fuel Manufacturing, may be exempted from participation in the training program described above upon the successful completion of a challenge examination. Individuals not familar with site specific information shall be instructed in those details to allow their safe conduct while at the facility.

2.5.2 Refresher Training

Employees and visitors (as necessary) working in the Nuclear Fuel Manufacturing facility Pellet Shop or whose job involves working with unclad nuclear material shall participate in an annual, not to exceed thirteen months, Radiation Worker refresher training program. The refresher training program shall emphasize the key safety aspects of their jobs and shall include, as a minimum: 1) a module covering significant abnormal occurrences and operational deficiencies identified at the facility and the corrective actions taken to preclude recurrence, 2) Radiation Safety, and 3) Criticality Safety.

2.5.3 Training Records

Formal training sessions shall be documented and competency demonstrated by passing a test to verify training effectiveness. When changes are made in radiation protection or criticality safety limits, affected individuals shall be informed and instructed in the new material. At the discretion of the Manager, Radiological Protection and Industrial Safety and based on the complexity of the new material, formal testing to assess an individuals understanding may be waived.

Training records shall be retained for the duration of an individuals employment at Combustion Engineering or a minimum of two years, whichever is greater.

2.6 Operating Procedures

Routine Nuclear Fuel Manufacturing facility operations which involve licensed materials shall be conducted in accordance with written procedures. The preparation of written safety related procedures are the responsibility of the cognizant Line Manager and shall be approved by the cognizant Manager, the Manager of Radiological Protection and Industrial Safety and the Nuclear Criticality Specialist. Written procedures that affect radiation and/or nuclear criticality safety shall include limits and controls which are required for nuclear safety of the subject activity.

The preparation, review, revision, approval and implementation of safety related operating procedures shall be accomplished through a document control system. As a minimum, safety-related operating procedures shall be reviewed every two years. This review shall be conducted by the cognizant Line Manager and individuals having the qualifications of the Radiological Protection and Industrial Safety Manager and the Nuclear Criticality Specialist in their respective areas of expertise.

Safety related operating procedures, and changes thereto, shall be retained for a period of six (6) months following procedure revision or termination of the operation involved, whichever is longer.

2.7 Internal Inspections and Audits

Inspections are routine reviews to check that operations are being conducted according to approved procedures. Audits are independent formal examinations made to verify that operations are being conducted according to established criteria. Audits are more formal and less frequent than inspections.

2.7.1 Inspections

The inspection function (assuring that operations are being conducted in compliance with regulatory requirements, license conditions, posted safety limits and safety related written operating procedures) is a normal part of the Radiological Protection and Industrial Safety Technician's job. Technicians are assigned to all operating shifts. As such, the inspection function is informally satisfied on an ongoing basis. On a monthly basis, Technicians shall perform a documented inspection using a prepared checklist to review the conduct of facility operations. Any time the Technicians find discrepancies, the cognizant Line Manager is informed of the remedial actions to be taken and a written deficiency report is turned over to the Manager, Radiological Protection and Industrial Safety. The Manager or his/her designee, assures that any necessary corrective actions specified are adequate and that they will be initiated in a timely manner. The Manager, Radiological Protection and Industrial Safety shall also ensure that quarterly fire safety, hazardous material safety (non-radioactive) and environmental protection inspections are performed by members of his/her staff. These inspections are also to be conducted with the aid of a prepared checklist. The Manager shall assure that any necessary corrective actions are initiated in a timely manner.

Completed inspection checklists shall be signed by the Technician performing the inspection and turned over to the Manager, Radiological Protection and Industrial Safety for review. A copy of the checklist is also provided to the Chairperson of the Facility Review Group.

Records of all deficiency reports and inspection checklists shall be retained for a period of at least three (3) years from the date of issue or until closure of all findings is obtained, whichever is later.

2.7.2 Audits

Audits are performed by individuals independent of day-to-day operating activities being audited at the Nuclear Fuel Manufacturing facility to verify that operations are being conducted according to established criteria. Audits are conducted in accordance with a written plan.

The Chairperson of the FRG shall ensure the conduct of the following audits in accordance with the schedule stated below:

Radiological Protection Program	Semiannually
Criticality Safety Program	Quarterly
Industrial Safety Program (including Fire & Chemical Safety)	Semiannually
Environmental Monitoring Program	Annually

Individual(s) conducting audits in the areas of radiological or criticality safety shall meet the qualification requirements of the Manager, RP&IS or a Senior Criticality Specialist, respectively. These audits may be conducted by members of the FRG, members of the Nuclear Fuel Staff, or consultants approved by the FRG Chairperson. A report of the audit findings shall be submitted to the Uranium Plant Manager for disposition of the findings, and to the FRG for review and trending.

Records of audit reports shall be retained for a period of three (3) years from the date of issue or until closure of findings is obtained, whichever is later.

2.7.2.1 Safety Committee Oversight

An independent Safety Committee shall conduct an annual audit for Nuclear Fuel Manufacturing operations involving licensed material covered by this application. The Safety Committee shall be appointed by and report to the President, Nuclear Fuel and be comprised of senior engineers and scientists from within C-E's nuclear technical community. The Committee may establish subcommittees and/or use consultants, as necessary, to carry out its responsibilities. Written audit findings shall be provided to the President, Nuclear Fuel ; Vice President, Manufacturing Operations; Uranium Plant Manager; Director, Product Development; and the Chairperson of the FRG .

Audit findings, as well as documentation of corrective actions shall be retained for a period of three (3) years from the date of issue or until closure of findings is obtained, whichever is later.

2.8 Investigations and Reporting

Abnormal occurrences are investigated in accordance with written procedures and are reported to the Uranium Plant Manager, Windsor Nuclear Fuel Manufacturing.

Reports to the Nuclear Regulatory Commission are made in accordance with specific conditions of this application and/or the applicable Federal Regulations. Regulatory Guide 10.1, Compilation of Reporting Requirements for Persons Subject to NRC Regulations, is used as a guide in identifying applicable reporting requirements. The level of investigation and the need for corrective action are determined based on the severity of the incident. The severity of an incident is based on the levels of uranium released and/or the degree of potential for exposure to workers or the public. An Abnormal Occurrence Review Committee is charged with the responsibility for investigating abnormal occurrences and recommending corrective action(s), as appropriate. The Committee is comprised of Line Managers from the Nuclear Fuel Manufacturing organization.

Records to investigations of abnormal occurrences reported to the Nuclear Regulatory Commission are retained for a period of three (3) years after closure of the investigation.

2.9 Records

Records pertaining to health and safety, facility modifications, abnormal occurrences, criticality analyses, inspections, audits, instrument calibrations, ALARA findings, employee training and refresher training, personnel exposures, routine radiation and contamination surveys and environmental surveys are retained to demonstrate compliance with the conditions of this application and the applicable Federal, State and local regulations. Records are retained for the periods specified in this application or the governing regulations, whichever is longer.

3.0 Radiation Protection

3.1 Special Administrative Requirements

3.1.1 ALARA Commitment

It is the policy of Combustion Engineering to conduct its business in a manner which ensures that its Nuclear Fuel Manufacturing Facilities are in compliance with radiation protection and other applicable regulations, and that the operation of these facilities will not be detrimental to the environment. In implementing this policy, Combustion Engineering shall ensure that radiation exposure to personnel (both in-plant and off-site) is maintained As Low As Reasonably Achievable (ALARA). In providing this assurance, conditions of applicable NRC and state licenses and other regulatory permits or licenses shall be complied with and regard shall be given to applicable NRC regulatory guides and industry standards.

For activities carried out within the scope of this application, responsibility for establishing and ensuring adherence to this policy shall rest with the Uranium Plant Manager, Windsor Nuclear Fuel Manufacturing. This policy shall be implemented through appropriate delegations to the Manager, Radiological Protection and Industrial Safety, and the applicable Line Managers.

3.1.2 Radiation Work Permit Procedures

All non-routine maintenance or repair operations on equipment involved with handling radioactive material within

Nuclear Fuel Manufacturing involving those non-routine maintenance operations in which ventilated containment systems are breached shall be covered by a Radiation Work Permit (RWP).

The RWP shall be requested by the cognizant engineer or supervisor and it shall establish the radiological safety requirements. The Manager, Radiological Protection and Industrial Safety shall approve the initial issue of all RWPs. Subsequent RWPs, for similar activities, may be re-authorized by the Supervisor, Radiological Protection and Industrial Safety. RWPs may be verbally authorized in circumstances where the qualified individual is not on-site. Such verbal authorization shall be documented and the subject RWP shall be signed within one working shift of the qualified individuals return to the site. Radiological Protection and Industrial Safety Technicians are responsible for monitoring proper implementation of RWP's.

As a minimum, RWP's shall be reviewed for their need every 30 days. The Manager or Supervisor of Radiological Protection and Industrial Safety acting on input from the Radiological Protection and Industrial Safety Technicians shall close out all RWP's to assure that the specific work was completed in a satisfactory manner prior to permitting restart of the subject operation.

3.2 Technical Requirements

3.2.1 Access Controls

All personnel ente. .ng the unclad fuel handling areas must do so through the change areas provided for this purpose.

As a minimum, the following protective clothing shall be worn:

- Coverall or Lab Coat
- Special shoes or shoe covers
- Safety glasses

Additional protective clothing which may be required for non-routine operations shall be prescribed by the applicable RWP in the Nuclear Fuel Manufacturing facility.

3.2.2 Personnel Monitoring Requirements

All personnel must wash their hands before exiting the contaminated area and, as a minimum, shall monitor their hands, exposed areas of the body and personal clothing with the alpha personnel monitor located at the change line.

The frequency and control levels of monitoring personal clothing and body surfaces shall be as follows:

PERSONAL CLOTHING AND BODY SURFACE ALPHA ACTIVITY CONTROL LEVELS

Surface	Alpha dpm/100cm2	Min. Survey Frequency
Personal Clothing, Body Surfaces (Hair, Face, Hands)	Indistinguishable from background	Before leaving contaminated area and when contamination is observed on body surfaces.

If indicated levels are greater than the control level, the individual shall promptly notify a member of the Radiological Protection and Industrial Safety staff and shall not leave the contaminated area until they respond and the situation is resolved.

3.2.3 Ventilation Requirements

Nuclear Fuel Manufacturing

Ventilation in the Nuclear Fuel Manufacturing facility (Building #17) is provided by four separate exhaust systems as follows:

FA-1 Powder Preparation and Pressing - This system has a capacity of 12,100 CFM and incorporates prefilters and a double bank of 12 absolute filters, each 99.97% efficient at 0.3 microns. The air exhaust from this system which is either returned to the inclad fuel area or released from the plant is sampled 100% of the time and analyzed each day.

FA-2 Furnace H2 Burnoff - This system has a capacity of 1340 CFM and incorporates prefilters and a single bank of 4 absolute filters, each 99.97% efficient at 0.3 microns. The air exhaust from this system is released from the plant and sampled 100% of the time and analyzed each day.

FA-3 Pellet Grinding and Rod Loading - This system has a capacity of 17,500 CFM and incorporates prefilters and a double bank of 16 absolute filters, each 99.97% efficient at 0.3 microns. The air exhaust from this system is released from the plant and sampled 100% of the time and analyzed each day.

FA-4 Recycle Powder Area - This system has a capacity of 6000 CFM and incorporates prefilters and a double bank of 6 absolute filters, each 99.97% efficient at 0.3 microns. The air exhaust from this system is released from the plant and sampled 100% of the time and analyzed each day.

The capacity of the ventilation systems have been matched to provide a negative pressure differential between the Pellet Processing Facility and all surrounding work areas. The direction of air flow shall be checked monthly and documented. If airborne activity results, averaged over a two week period, exceed 25% of the applicable concentration listed in Table II, Column I of 10 CFR 20

Appendix B for air being discharged to an unrestricted area (from manufacturing), an investigation will be conducted and corrective actions taken. In addition, to assure our releases remain as low as reasonably achievable, a quarterly limit of 18 uCi in gross alpha activity of total uranium in plant gaseous effluents shall be maintained. If the radioactivity in plant gaseous effluents exceeds 18 uCi, a report which identifies the cause for exceeding the limit and the corrective actions to be taken to reduce release rates shall be submitted to the NRC within 30 days. Also, if the parameters important to a dose assessment change, a report shall be submitted within 30 days which describes the changes in parameters and includes an estimate of the resultant change in dose commitment. The 18 uCi/qtr limit would result in a lung dose to an individual at the nearest residence of (conservatively) less than 0.10% of the 25 mrem/year standard as specified in 40 CFR 190.

Ventilation system filters and/or prefilters will be changed, rotated, or knocked down whenever a pressure drop of 4 inches of water is measured across the combination of the prefilter and first bank of absolute filters. The pressure drop for all 4 systems shall be checked weekly and documented. When the face velocity at a ventilated hood drops below 100 fpm, the hood filters or ventilation system filter will be changed, brushed, or knocked down to increase the air flow to 100 fpm minimum or the hood shall not be used to handle radioactive material. Face velocities will be checked weekly in the Manufacturing facility.

Any work on filter change involving any of the four fixed air systems in manufacturing shall be performed under an RWP. Following all filter changes or other movement of filters, both the Supervisor and a Technician from the Radiological Protection and Industrial Safety group shall inspect the placement of the absolute filters for proper sealing. In addition, air samples will be taken and counted immediately after 1/2, 2, and 8 hours of operation to assure the absolute filters are adequately filtering the exhaust air. The adequacy of the sampling techniques to obtain representative samples will be verified quarterly in the Manufacturing facility.

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3.2.4 Instrumentation

Capabilities of radiation detection and measurement instrumentation shall be as follows:

Alpha counting System	10 - 10,000 dpm
Alpha Survey Meters	0 - 50,000 counts per minute
Beta Gamma Survey Instruments	.05 mR/hr - 200 mR/hr
Neutron Survey Instruments	.5 - 5,000 mrem/hr

A sufficient number of the instruments, meters and systems listed above shall be maintained operational to adequately conduct our Radiological Protection and Industrial Safety program. Additional instrumentation is maintained for emergency use as outlined in Part I, Section 8. The detectors for the criticality alarm system are calibrated semi-annually and following any repair that affects the accuracy of the measurements. All other instruments are calibrated semi-annually and following any repair that affects the accuracy of the measurements. The calibration of the survey instruments shall meet the specifications described in Section 1.11 of

Regulatory Guide 8.24, "Health Physics Survey During Enriched Uranium-235 Processing and Fuel Fabrication". The alpha counting equipment is checked daily to verify background and efficiency.

3.2.5 Internal Exposure (Fixed Location Breathing Zone Air Sampling and Fixed Location General Air Sampling

Nuclear Fuel Manufacturing

The room air in all areas where unclad licensed material is handled, processed, or where operations could result in worker exposure to the intake of quantities of uranium exceeding those specified in 10 CFR 20.103, shall be continuously sampled and analyzed on every shift for radioactivity. Air sampling shall be accomplished using fixed-location general air sampling stations and fixed location breathing zone air sampling stations. The sampling results from the fixed location breathing zone air sampling stations shall be used for the basic evaluation of the internal exposure of workers. During the normal operating period, if a single breathing zone or general air sampling station indicates the airborne concentration of radioactivity for that area exceeds the MPC specified in Table I Column I of 10 CFR 20, Appendix B, an investigation as to the cause shall be conducted. Any necessary corrective actions to prevent its recurrence shall be taken and documented. Both fixed location breathing zone and general air samplers shall have a minimum flow of 10 lpm.

The fixed locations breathing zone air sampling stations shall be the primary means of determining radioactive airborne concentrations of particulates in the workers breathing zone. The fixed location breathing zone air sampling stations shall be strategically located throughout the shop and run continuously during operations. The fixed location general air sampling stations provide air sample which are representative of working areas in order to verify adequate ventilation and contamination control. All samples from both fixed location breathing zone and general air sampling stations shall be analyzed after each working shift.

The representativeness of the fixed location breathing zone air sampling stations shall be evaluated at lease once every 12 months and whenever any licensed process or equipment change is made.

It is recognized that the behavior of individual operators can be a significant contributing factor to an individual's exposure, and that this may not be amenable to the desired degree of improvement. Where the individual operator is found to contribute significantly to higher exposures, closer personnel surveillance shall be maintained. An individual whose 40 hr. exposure exceeds 2.5 MPC days shall be closely monitored with a portable BZ sampler. The sampler will have a minimum flow rate of 1400cc/min. If the person's 40 hr. exposure exceeds 4 MPC days, the person shall be removed from exposure to airborne contamination. It is the responsibility of the Supervisor, Radiological Protection and Industrial Safety to evaluate these situations to determine the relative contributions of individuals and equipment. The pellet shop is the only place in the manufacturing facility that handles unclad UO2. This portion of the facility is kept at a negative pressure as described in Section 3.2.3. therefore, continuous air sampling shall be conducted in this area only.

3.2.6 External Exposure (Dosimetry Requirements)

Each individual who enters a restricted area under such circumstances that he is likely to receive a dose in any calendar quarter of 25 percent of the applicable value specified in 10 CFR 20.101(a) shall be supplied with a TLD badge and indium foil for purposes of personnel dosimetry. Badges will be processed monthly. When a high exposure is suspected, the individual's badge will be sent out for immediate processing. All visitors will be supplied with indium foil badges. Area TLD badges and neutron foils are also strategically placed throughout the facility for the purpose of recording background radiation levels as well as radiation resulting from a criticality accident. The TLD badges will also be processed monthly during normal operations and immediately following a criticality accident. Procedures to determine high radiation doses immediately following a criticality accident are described in the Emergency Procedures Manual.

3.2.7 Bioassay Program

Urinalysis and in-vivo lung counting shall be conducted annually. Urine samples for any workers who routinely work with soluble compounds of uranium shall be collected and analyzed monthly. If the most recent quarterly average of the airborne uranium concentration for any work area exceeds 25% of the respective DAC, the frequency of sampling and the type of bioassay measurements for workers in that work area shall be modified to that given in Table 3 of Regulatory Guide 8.11, "Application of Bioassay for Uranium". The following tables outline the action levels which will be utilized for both urinalysis and in-vivo counting:

URINALYSIS ACTION LEVELS

Urinalysis Result	Action
1) Sample >25 ugU/liter	1) Confirm result (if unexpected) 2) If result is confirmed: - Impose work restrictions - Collect and evaluate diagnostic urine samples - Conduct investigation to identify probable cause - Perform In-Vivo count

IN-VIVO ACTION LEVELS

In-Vivo Result	Action
1) Lung Burden >175 ug U235	1) Confirm result (if unexpected) 2) If result is confirmed: - Impose (or continue) work limitations. - Conduct job investigation to identify probable cause, determine if others were exposed, and evaluate adequacy of air sampling. - Initiate corrective actions.

3.2.8 Contamination Surveys

3.2.8.1 Contaminated Areas (Pellet Shop Building #17)

Removable Alpha Contamination	Action to be Taken
10,000-dpm/100 cm2	Immediate Clean-Up
5,000 dpm/100 cm2	Clean-Up within 24-hours

Contaminated areas shall be surveyed on a weekly basis. Material fixed on processing equipment or on surfaces shall be limited as required to control airborne radioactivity and external radiation exposures.

3.2.8.2 Clean Areas (Other plant areas, office areas, lunch areas)

Removable Alpha Contamination	Action to be Taken
Alpha Level	
100 dpm/100 cm2	Immediate Clean-Up
50 dpm/100 cm2	24-hour Clean-Up
10 dpm/100 dm2 (lunch rooms only)	Immediate Clean-Up

Other manufacturing areas, office areas, and the warehouse (Bldg. 21) shall be surveyed on a monthly basis. The lunch rooms shall be surveyed once a day, as a minimum.

Fixed Alpha Contamination

Monthly fixed alpha contamination levels in the non-contaminated areas (and for release of equipment from contaminated areas) shall be less than 500 dpm/100 cm2 average.

3.2.8.3 Materials & Equipment Released for Unrestricted Use (does not include the abandonment of Buildings)

The release of materials and equipment for unrestricted use shall be in accordance with "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material", USNRC, Annex B, July 1982.

3.2.9 Respiratory Protection

The Respiratory Protection Program shall be conducted in accordance with the USNRC Regulatory Guide 8.15.

4.0 Nuclear criticality Safety

4.1 Administrative Requirements

4.1.1 Double Contingency Policy - Process designs shall, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

4.1.2 Written Procedures and Approval Authority - All process operations involving SNM shall be covered by a shop traveler and/or an operation sheet which shall be followed. Precautions and limits regarding criticality and radiological safety shall be included in these procedures. In addition, all procedures shall provide for the labeling of mass limited containers to indicate the enrichment and the uranium content. All process equipment and operating areas shall be labeled to indicate the enrichment. Geometry limited containers will be handled as though they are full unless specifically labeled empty. Labeling shall be carried out under the direction of the cognizant supervisor.

These procedures shall be approved by the Manager, Radiological Protection and Industrial Safety. However, procedures involving a change in the criticality safety controls used for that particular process in the past shall also be reviewed and approved by the Nuclear Criticality Specialist. It is the responsibility of the Uranium Plant Manager to ensure that personnel are trained in operations and the corresponding safety limits.

It shall be the responsibility of the Supervisor, Radiological Protection and Industrial Safety to assure that each work station is properly posted.

4.1.3 Request for Changes and Criticality Analysis

All proposed changes in process, equipment, and/or facilities that could affect nuclear criticality, radiological or industrial safety shall be approved in accordance with the responsibilities and authorities set forth in Section 2.1 of this part. The necessary analysis and resultant safety limits shall be established by a person having the minimum qualifications of a Nuclear Criticality Specialist. Procedures have been established for requesting changes and all request forms, approval forms, and associated documentation shall be maintained under the direction of the Manager, Radiological Protection and Industrial Safety.

4.1.4 Posting of Limits

All special nuclear material work stations and storage areas shall be posted with a nuclear criticality safety limit approved by the Manager of Radiological Protection and Industrial Safety and the Nuclear Criticality Specialist. The Supervisor, Radiological Protection and Industrial Safety, maintains records of the review and approval of each posted nuclear criticality safety limit.

4.1.5 Internal Review Requirements

All process/equipment/facility changes which affect nuclear criticality safety shall be reviewed and approved in writing by the Nuclear Criticality Specialist and the Manager, Radiological Protection and Industrial Safety. An independent review and written approval shall be performed by the Senior Nuclear Criticality Specialist.

As stated in Section 4.1.3, all such approvals shall be recorded in a log maintained under the direction of the Supervisor, Radiological Protection and Industrial Safety.

4.1.6 Marking and Labeling of SNM - All mass-limited containers shall be labeled as to enrichment and content. All geometry limited containers and processes are safe up to the maximum allowable enrichment of 5.0wt% U235.

4.1.7 Section Deleted

4.1.8 Section Deleted

4.2 Technical Requirements

4.2.1 Preferred Approach To Design - It is the intent of Combustion Engineering to use physical controls and permanently engineered safeguards on processes and equipment in the establishment of nuclear safety limits wherever practical. Use of administrative controls in the establishment of safety limits will be minimized.

4.2.2 Basic Assumptions and Analytical Methods - Written health and safety restrictions for all operations on radioactive material shall be provided in the form of approved Radiation Work Permits or approved detailed procedures, and appropriate operational limits shall be posted in the vicinity of work stations in the manufacturing facility.

shall be separated from any other fissile material by 12 feet. Rods containing sintered UO2 pellets enriched to a maximum of 5.0% U235 shall be stored in Building #5. Slab storage in Building #5 shall be in accordance with Table 4.2.5.

Criticality safety of the less complex manufacturing operations is based on the use of limiting parameters which are applied to simple geometries. Safe Individual Units (SIU) shall be selected from Table 4.2.5. These units shall be spaced using the surface density method.

The remaining manufacturing operations are evaluated using two dimensional transport and/or 3 dimensional Monte Carlo Codes. The sixteen group Hansen-Roach cross section library is used for homogeneous systems, while NITAWL and XSDRNPM are used to generate multigroup cross sections for heterogeneous systems. All calculational methods involving computer codes shall be validated in accordance with the criteria established in Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities."

4.2.4 Safety Margins - Individual Units - Safety margins applied to units calculated to be critical (with up to 2% uncertainty), and incorporated in the SIU's shall be as follows:

Mass	2.3
Volume	1.3
Cylinder Diameter	1.1
Slab Thickness	1.2

These values shall be further reduced where necessary to assure maximum fraction critical values of 0.4 for geometrically limited units, and 0.3 for mass limited units (based on optimum water moderation). An additional reduction has been applied to several mass and volume limits to assure that spacing requirements remain constant for all enrichments. For validated computer calculations, the highest K eff for a single unit or array shall be 0.95 including a 2-sigma statistical uncertainty and including all applicable uncertainties and bias.

The basic assumptions used in establishing safe parameters for single units and arrays shall be as follows:

- The possibility of accumulation of fissile materials in inaccessible locations shall be minimized.
- Nuclear safety shall be dependent on the degree of moderation within the process unit. Additional moderating materials, when considered to be credible, will be included in the analysis.

- Optimum conditions of water moderation, reflection, and heterogeneity for the system shall be determined in all calculations.
- The analytical method(s) used for criticality safety analysis and the source of validation for the method(s) shall be specified.
- Safety margins for individual units and arrays shall be based on accident conditions such as flooding, multiple batching, and fire.

4.2.4 A Moderation Control

- For moderation control a maximum K_{eff} of 0.95 shall apply for validated computer calculations. The 0.95 K_{eff} value shall be reduced by (1) the applicable 2 sigma statistical uncertainty associated with Monte Carlo calculations and (2) the applicable uncertainties and bias associated with the bench marked calculations.

  The basic assumptions used in establishing safe parameters for single units and arrays shall be as follows:

- Nuclear safety shall be independent of the degree of moderation between units up to the maximum credible mist density of 0.1% H_2O (0.001 gm H_2O/cc).

- Criteria used in the choice of fire protection in areas of potential criticality accidents (when moderators are present) shall be justified in writing. An audit of the existing fire sprinkler system in building 17 (Figure 4.2.3) shall be conducted once a quarter (Sprinkler Heads, Risers, Distribution Lines, and Pumps) to see to it that it had not been modified or added to in any way that would impair its performance or have an effect on calculated mist density. All proposed changes to the fire sprinkler system, that could affect building 17 will be reviewed and approved in accordance with Section 2.1.6, as regards facility changes affecting criticality, for their effect on mist density before such changes are implemented.

- Plastic bags which are placed around the fuel assembly shall be left open at the bottom at all times including the period in which the assembly is in storage.

  Combustible materials in the area shall be minimized at all times.

- In any area where unsealed powder or randomly loaded pellet containers are exposed to the fire sprinkler system they will be assumed to fill with water. Hard scrap (pellets or pieces thereof) shall be assumed as optimally moderated.
- Possible moderating material around fissile materials will be included in the analysis.

4.2.5 Limits for Safe Individual Units (SIU's)

Table 4.2.5

Type of Limit	Maximum Limit
Pellet Shop	
Mass (powder or randomly looded pellets in ≥ 5 gal steel containers)	35.0 kg ≤ UO_2 ≥5.0% U235
Coplanar Slab (Powder/Pellet) (Pellets Randomly Loaded) (Sintered pellets up to 0.4" diameter) (Green pellets up to 0.43" diameter)	4.0" Maximum Height.
Coplanar Slab (Hexagonally stacked rods) (Sintered pellets up to 0.4" diameter)	6.0" Maximum Height.
Hard Scrap	16.0 kg UO_2 ≤5.0% U235
Cold Shop	
Slab (Hexagonally stacked rods)	6.0" Maximum Height.
Fuel Assemblies	keff ≤ 0.90
Pre-stacked Array	keff ≤ 0.90

Building 21

Storage of SNM in approved shipping containers
(TI=80 per array)

4.2.6 Interaction Criteria - Activities involving SNM may be conducted in single or two level areas of the facility.

All mass units shall have a separation of at least one foot, edge to edge.

Spacing for mass unit activities carried out in the single level portions of the facility shall be such that the contained UO2 and moderator, if "smeared" over the allowed spacing areas would not exceed 50% of the critical water-reflected infinite slab surface density assuming optimum water moderation for minimum mass per unit area. Co-planar slabs specified in Table 4.2.5 require no additional spacing if on the same plane. Non-co-planar slabs within 4 feet of each other are limited to a maximum of 12-inch vertical differences, and must be separated by a 12-inch minimum horizontal spacing.

Portions of the facility contain two levels, each of which may be used for SNM. Mass limits on each level shall be spaced such that the contained UO2 and moderator, if "smeared" over the allowed spacing areas would not exceed 25% of the critical water-reflected infinite slab surface density assuming optimum water moderation for minimum mass per unit area.

All array calculations have been performed assuming a doubly infinite planar system, based on the consideration that components of subcritical infinite arrays can be combined where the unit size and cell spacing is preserved. Array reflection consists of a 16" concrete floor, and a 4" thick concrete roof 25 feet above the floor.

Table 4.2.6

The spacing requirement for mass SIU's specified in Table 4.2.5 is shown below. Spacing areas shall be established to provide equal distances from the edges of the units to the spacing boundary in all directions.

Limit	Spacing Areas
Mass	6.73 ft2

Justification for this spacing criteria is provided in Part II of this application.

Whenever more than one mass SIU is allowed in any given hood or box, positive spacing fixtures shall be used to assure spacing. Carts, limited to one mass SIU shall measure at least 2.6 feet on a side, and shall be designed to assure that the Mass SIU is centered.

In cases where the spacing area extends beyond the equipment boundaries, such as the storage facilities, the spacing boundary shall be indicated with a colored line. The line may be crossed by carts only to permit an operator to transfer that SIU to an available storage position.

4.2.7 Structural Integrity Policy - All storage racks, furnaces, containment, and processing equipment which provide nuclear safety limiting parameters shall be designed to assure agains failure under normal and reasonable overload conditions and under conditions of shock or collision foreseeable in the plant area. All equipment designed shall incorporate a minimum safety factor of 3.0. All equipment design shall conform to standard design practices, thereby assuring adequate structural integrity. Materials of construction shall be selected to assure, as far as possible, resistance tc fire and corrosion. The individual engineer responsible for the purchasing or design of the new equipment shall assure that the minimum safety factor of 3 has been incorporated intc the design of the equipment. The minimum qualifications for engineers shall be a bachelors degree in engineering or related fields.

4.2.8 Zoning for Fire Protection - An overhead sprinkler system as well as portable extinguishers are located throughout the fuel manufacturing facilities Onsite and Offsite fire protection service personnel have been instructed to use only portable dry chemical extinguishers in the Bldg. #17, to maintain the highest possible margin of nuclear criticality safety. Fire hoses shall not be permitted in Bldg. #17.

4.2.9 Criticality Alarm System - A criticality alarm system which meets the requirements of 10 CFR 70.24 (a) (1), Regulatory Guide

8.12, "Criticality Accident Alarm system" shall be maintained in the manufacturing facility, in accordance with requirements of 10CFR70.24(a). the detectors operate in the range of 1-10,000 mR/hr. The locations of the detectors within the manufacturing facility are shown on Page II.8-75. The radiation exposure rate is shown on a central panel located in the Building 17 entrance area for Buildings 17 and 21. There is an alarm which serves as a local and general audible radiation evacuation alarm. When the alarm is sounded, the Emergency Plan is immediately put into effect. The monitors are connected to the emergency power system, which is supplied to all emergency lights and alarms in the event of a general power failure within the facility. This electrical system renders the alarm system operative at all times.

Alarm operational tests of radiation monitors are performed monthly by Radiological Protection and Industrial Safety Personnel. A radioactive source is used to perform these test. The entire system is tested semiannually and following any repair that may affect system performance.

4.3 Specific Criticality Safety Criteria

Specific criticality safety criteria in addition to the general criteria described in Section 4.2 are necessary to assure nuclear safety for several process operations, as described below:

4.3.1 All incoming UO2 powder shall be stored in 9.75" diameter x 11" long stainless steel cans. All powder shall be sampled before being placed in the virgin powder storage area to demonstrate on a 95/95 confidence level that the moisture content of powder lots is less than 5.0 wt.%. In addition, all damaged packages where containment is breached shall be sampled. The area in and around the virgin powder storage area shall be kept free of combustibles.

4.3.2 The fire door on the virgin powder prep storage area shall close automatically on activation of the fire alarm or upon electrical power failure. The automatic closing feature of the door on the virgin powder storage area shall be verified quarterly and records of its performance shall be maintained.

4.3.3 A maximum of three 9.75" diameter x 11" long stainless steel powder containers and one 5 gallon powder container or two 9.75" diameter x 11" long stainless steel powder containers and two 5 gallon powder containers shall be allowed in the batch make-up hood in the position shown in Figure 8.2.

4.3.4 The one 5-gallon pail being filled from the three other containers in the batch Make Up Hood shall be limited to 35 Kg UO2 and shall be sealed with a water tight cover prior to being stored on the conveyor.

4.3.5 The blender hoods shall be restricted to 35 Kg UO2 per hood. This does not include the UO_2 in the transfer tube which was assumed to be full of UO_2 in the analysis.

4.3.6 The wiper blade, powder plenum, and the drying belt at the Powder Preparation Station shall be inspected once per week to assure that the wiper blade is functioning properly and that no fuel is accumulating in the plenum below the belt. The depth of the powder on the drying belt is limited mechanically to 1/2" thickness. The drying belt shall be completely enclosed. The powder accumulation under the drying belt shall be less than 1/2".

Records of these inspections shall be maintained.

4.3.7 In the Concrete Block Storage Area, A maximum of 35.0 Kgs UO2 may be contained in 5-gallon or smaller containers. Each storage position shall be limited to one container.

4.3.8 UO2 pellet thickness on each of the Pellet Storage Shelves shall meet the slab limit specified in Table 4.2.5. The shelves shall be covered from above by a sheet metal top.

4.3.9 Storage of sintered pellets shall be limited to the slab limit specified in Table 4.2.5.

4.3.10 Touching clad rods in horizontal storage shall be close packed in a hexagonal lattice and shall meet the slab limit specified in Table 4.2.5.

4.3.11 A maximum of 32 fuel rods shall be allowed in each autoclave.

4.3.12 The boxes on the Double Shelf Rod Storage Racks shall be covered with a tight fitting aluminum cover which overlaps the outside edge of the box by a minimum of one inch. Fuel rods shall be close packed in a hexagonal lattice and shall meet the slab limit specified in Table 4.2.5 within the individual rod boxes. One box may remain uncovered for short periods of time to allow for the addition or removal of rods for inspection purposes provided that personnel are in attendance. Boxes shall be a minimum of 6 inches edge-to-edge both vertically and horizontally. The center-to-center distance between adjacent racks shall be at least 55 inches.

4.3.13 In the Rod Storage Area, fuel rods in each fuel rod storage box shall meet a slab height limit of six (6) inches. Since the Rod Storage Area is dry, the fuel rods may be stored in any array.
The entire storage array is covered by a fire resistant roof to assure the exclusion of sprinkler water. Large signs are posted over the storage array that say "Do Not Use Fire Hoses in this Area."

4.3.14 Fuel assemblies shall be stored only in positions described in Figure 8.11, Part II. The assemblies in the storage positions only shall be wrapped with polyethylene with the bottom ends open to assure drainage. Fire fighting in the assembly storage room with fire hoses is prohibited.

4.3.15 Shipping containers, each containing 2 fuel assemblies, shall be stored outdoors in arrays up to three high. Containers shall be stored on pavement or blacktop within an 8 foot high chain link fence.

4.3.16 Waste containers shall be stored in designated areas of the pellet shop, on a concrete pad contiguous to the south wall of the Bldg. #21 warehouse, or in the temporary storage trailer located inside the Building 17/21 security fence. Packages will contain less than 350 grams U235 each which will meet the surface density criteria. Maximum residence time for packages stored on the pad shall be twelve months.

4.3.17 Incoming virgin powder in CE-250-2 shipping containers (Certificate of Compliance No. 9022) shall be stored in the Bldg. #21 warehouse or in the truck unloading area in the northwest corner of Building 17 in their original shipping containers only. The size of the array of the containers shall be in accordance with the requirements specified in the NRC Certificate of Compliance and all DOT regulations.

4.3.18 Incoming pellets in the UNC-2901 shipping containers (Certificate of Compliance No. 6294) shall be stored either within the transport vehicle or inside Building 17/21 complex. If stored within the transport vehicle said vehicle shall be inside the Building 17/21 security fence. Two UNC-2901 shipping containers are strapped to a pallet. Three pallets can be stored in the Building 17 Pellet Shop Annex and four pallets can be stored in the Building 17 Pellet Loading Area. The containers are received in a horizontal position. This position will be maintained when inside the Building 17/21 complex and the pallets shall be at least 1 foot from process equipment in the area.

4.3.19 The size of any array of shipping containers, with the exception of the 927A1 and 927C1 Fuel Bundle Shipping containers, shall be limited to a total transport index of 80. The 927A1 and 927C1 Fuel Bundle Shipping Container arrays shall not be more than three high. Shipping container arrays of different types shall be separated from one another by at least 20 feet.

4.3.20 All storage containers of UO_2 5 gallons or less located outside of hoods or in storage spaces shall be covered. Any storage containers accidentally internally moderated shall be handled as individual mass units and stored in the concrete block storage area.

4.3.21 The UNC-2901 Shipping Containers mounted on the shipping pallet can be opened only one at a time when located in an area free of other fissile material. This area shall be at least 21 ft^2.

4.3.22 The filled press feed hoppers can only be stored or placed in designated areas. Only one filled press feed hopper can be in transit on the pellet shop main floor and one can be in transit on the press feed mezzanine.

4.3.23 The maximum internal volume of the centrifuge shall be 22.0 liters. Other fissile material shall be separated from the centrifuge by at least one foot.

4.3.24 The maximum internal thickness of the slant (storage) tank for the centrifuge system shall be $\leq$4.15 inches.

4.3.25 The centrifuge and dump tank waste contents shall be isolated from the slant tank when the contents of the slant tank are diverted to Building 6. When the dump tank, centrifuge and slant tank circuit is operative, the discharge line to Building 6 shall be isolated.

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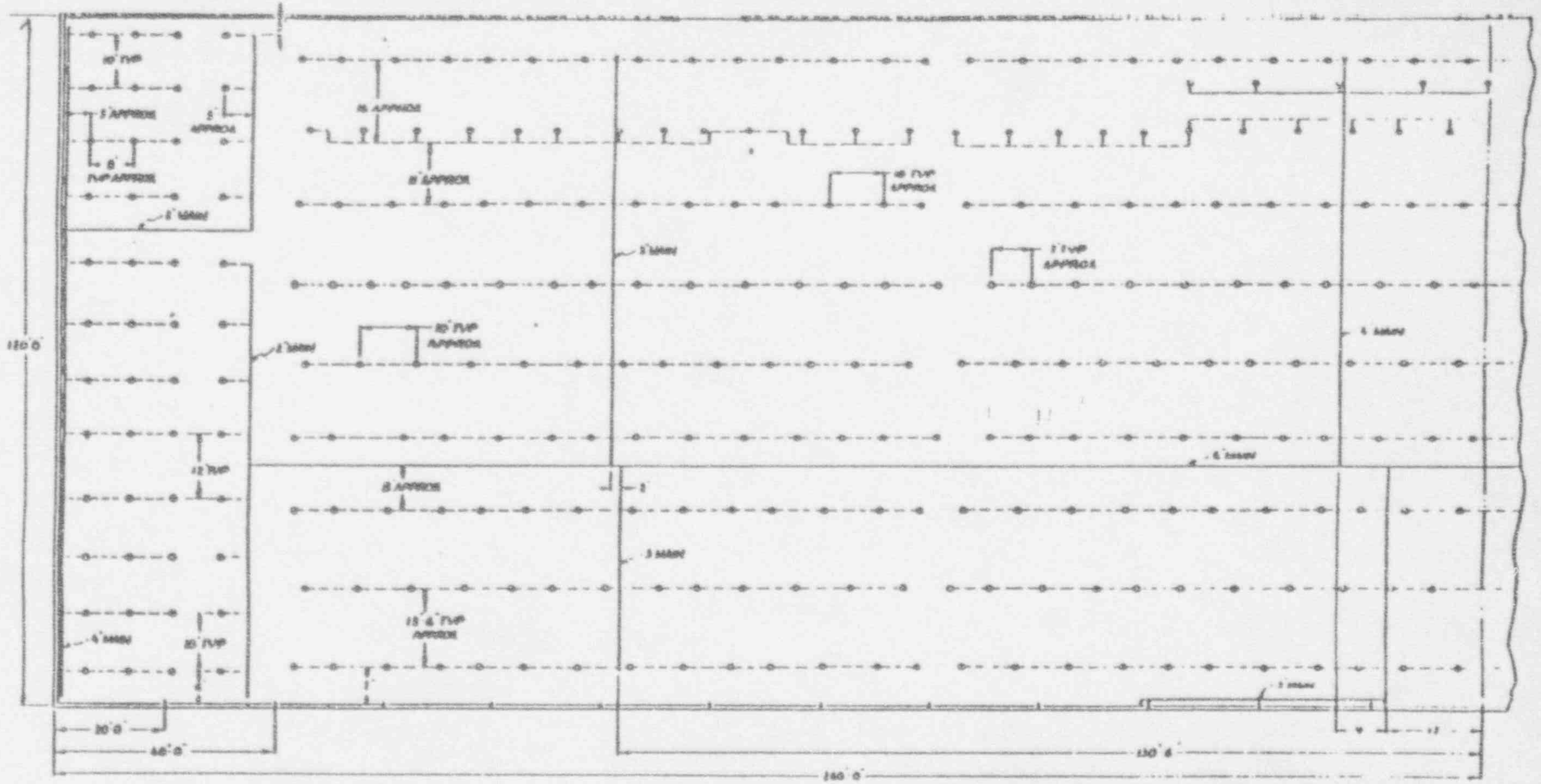

1. UNLESS OTHERWISE SPECIFIED ALL DIMENSIONS ARE IN FT

2. ALL SPRINKLERS ARE RELIABLE AUTOMATIC MODEL G, 1/2 STANDARD ORIFICE UPRIGHT UNIT

3. —— SIGNIFIES MAIN WATER FEED LINE

4. - - - SIGNIFIES SPRINKLER DISTRIBUTION LINE

5. SPRINKLER SYSTEM SUPPLIED BY 250,000 GALLON ELEVATED STORAGE TANK. PRESSURE ON SYSTEM IS MAINTAINED BY A MARATHON ELEC PUMP (MODEL [illegible]) BACKED UP BY A GM DIESEL PUMP (MODEL PTR 15D-50) AND TWO PEERLESS PUMPS (MODEL 15A11L3-D)

LICENSE No SNM-1067
DOCKET 70-800

FIGURE 4.2.3
LAYOUT OF SPRINKLER SYSTEM
REVISION 2

DATE: 6/15/93
PAGE: 1.4-19

5.0 ENVIRONMENTAL PROTECTION

5.1 Effluent Control Systems Commitments

5.1.1 Sanitary Wastes

All sanitary wastes are processed through an on-site sewage treatment facility to assure compliance with all state efflue discharge regulations.

5.1.2 Control of SNM-Bearing Effluents and Wastes

During the processing of Source and Special Nuclear Materials into fuel elements and fuel assemblies, a certain amount of scrap and waste is generated. Salvage of these materials, or discharge of them as wastes with non-recoverable uranium, will be carefully controlled. Water utilized for cleaning or processing in the unclad fuel handling area is centrifuged anc sampled before being discharged to holding tanks where the wat is again sampled before final discharge.

CE has available laboratories to perform analytical services a required, to determine uranium content and isotope ratios. These Laboratories are capable of the following:

- Wet chemical or gamma pulse height analyses for uranium content and wet chemical analyses for iron, nickel, chromiu cobalt, tin, boron, and other elements of interest.
- Emission spectrographic analyses for isotopic content anc procedures for neutron absorbing impurities, and solution ar oxide excitation techniques for impurities in alloying and cladding materials.
- X-ray fluorescent spectrometry on solids, liquids, and

powders for uranium content.

- Fluorometric analyses for measuring trace contaminants, such as uranium and beryllium and determining their content in wastes.
- Combustion techniques for carbon, hydrogen, and oxygen in uranium-containing bodies and cladding.
- Spectrophotometric determination of uranium content of salvage materials.

5.1.3 Low-Level Radioactive Waste

Low level radioactive wastes will be packaged in accordance with all applicable regulations and delivered to a carrier for transport to an approved disposal facility. Current copies of disposal facility regulations shall be maintained.

An outside pad has been approved for the storage of low level wastes. The pad is 14' x 80' and is contiguous to the south wall of the Building #21 warehouse. This pad is contained within an 8' high chain link fence.

Records are maintained to assure that no single package will contain more than 350 grams of U235. All packages will be placed on pallets and package stacking will be limited to two high. Maximum residence time of a package on the pad will be twelve months. Packages will not be opened outside the building under any circumstances. Packages will contain no liquid wastes.

In addition, all packages will be sealed, monitored for contamination and labeled as to enrichment and U235 content. All outside storage will be checked four times each year at which time contamination levels will be evaluated, and the adequate condition of the packages will be verified.

5.1.4 Liquid Wastes

All liquid wastes, in-process, and clean-up rinse water solutions are sampled to verify that MPC is not exceeded, and are then introduced to the liquid waste system as described below. Release of liquid waste will be authorized by a member of the Radiological Protection and Industrial Safety staff.

Sinks and showers in the manufacturing facility are drained to any one of ten (10) 2000-gallon retention tanks. The tanks fill automatically in sequence. When eight tanks become filled to capacity, a blinking warning light located in the outside wall of the building is activated to warn that two retention tanks remain in reserve to receive radioactive liquid waste before overflow might be expected. a sampling station is provided at the base of each retention tank. A 500-ml sample is withdrawn and forwarded to the Radiochemistry Laboratory for gross alpha and beta analyses. Water is discharged to the Windsor site creek which flows into the Farmington River at, or below 0.000003 uCi/ml (this is ten percent of MPC for insoluble natural uranium). The discharge level for unidentified mixtures of radionuclides is 0.000000003 uCi/ml. (This is ten percent of MPC for unidentified mixed radionuclides). Where levels of activity exceed these limits the water is diluted before being discharged. The instruments measuring the liquid-waste level in each dilution tank shall be calibrated on an annual basis.

5.1.5 Airborne Effluents

Airborne Effluents shall be continuously monitored in accordance with Section 3.2.3 and discharge to the environment shall be as specified therein. If the radioactivity in plant gaseous effluents exceeds 18 uCi gross alpha activity of total uranium per calendar quarter, a report to the NRC shall be prepared and submitted within 30 days which identifies the cause for exceeding the limit and the corrective actions taken to reduce the release rates.

5.2 Environmental Monitoring Program

5.2.1 Fallout Stations

Ten stations for collecting rainfall and particulate fallout are distributed concentrically in the Windsor site property. These samples shall be analyzed for gross beta and gross alpha radioactivity and total uranium.

5.2.2 Liquid Samples

Liquid Samples shall be taken from the site wells and ponds, the site creek and points upstream and downstream from the confluence of the site creek and the Farmington River. These samples shall be analyzed for gross beta and gross alpha radioactivity, pH, nitrates, fluorides and total uranium.

5.2.3 Sediment

Sediment samples shall be taken from the site ponds, the industrial stream and points upstream and downstream from the confluence of the industrial stream and the Farmington River.

The samples shall be analyzed for gross alpha and gross beta radioactivity and total uranium.

5.2.4 Vegetation and Soil Samples

Vegetation and soil samples shall be collected at each of the fallout station locations on-site and four locations in the grassy area surrounding Building #17.

Additional samples are collected off-site in the tobacco fields north, south, east and west of the site boundary.

These samples shall be analyzed for gross alpha and gross beta radioactivity and total uranium.

5.2.5 Sampling Schedule

The above environmental monitoring program shall be carried out in accordance with the schedule provided in Table 5.1. All sample results shall be documented.

TABLE 5.1

CE ENVIRONMENTAL MONITORING PROG-AM

SAMPLE	FREQUENCY	LOCATION	ANALYSIS	VOLUME
1. Farmington River Surface Water, Industrial Stream and Site Ponds	Quarterly in March, May, August and November	Four locations on the Farmington River, the site ponds and Industrial Stream	Gross Alpha and Beta, Nitrate, Flouride, pH, Total Uranium	1.25 liters
2. Well Water	Quarterly in March, May, August and November	Each site well	Gross Alpha and Beta, Nitrate, pH, Flouride, Total Uranium	1.25 liters
3. Sediment from Farmington River, Site Ponds and Industrial Stream	Quarterly in March, May, August and November	Same locations as Surface Water	Gross Alpha and Beta, Total Uranium	One pint
4. Vegetation On-site	Semi-annually in May and September	Each Fallout Station Location and four locations in grassy areas surrounding Building #17	Gross Alpha and Beta, Total Uranium	One pint of packaged vegetation
Off-site	Semi-annually in May and September	Tobacco fields on North, South, East and West Site Boundary	Gross Alpha and Beta, Total Uranium	One pint of vegetation, tobacco leaves at end of harvest
5. Soil	Semi-annually in May and September	Same locations as vegetation	Gross Alpha and Beta, Total Uranium	One pint (Upper inch)
6. Fallout	Quarterly in March, May, August, and November	Ten locations on Site	Gross Alpha and Beta, Total Uranium	Total Continuous Collection

6.0 INDUSTRIAL SAFETY

The Manager, Radiological Protection and Industrial Safety shall be responsible for defining all programs and standards related to Industrial Safety, including OSHA regulations, for all activities in the Nuclear Fuel Manufacturing Facility. The Industrial Safety Specialist, reporting to the Manager, Radiological Protection and Industrial Safety, is responsible for implementing those programs and standards. The Radiological Protection and Industrial Safety Technicians monitor the day-to-day compliance.

7.0 DECOMMISSIONING PLAN

Combustion Engineering's Decommissioning Plan dated January 15, 1979 was submitted previously and is included as Appendix A to this license.

8.0 RADIOLOGICAL CONTINGENCY PLAN

Combustion Engineering's Radiological Contingency Plan, approved as Amendment No. 35 to SNM-1067 on March 26, 1982, is considered to be part of this license.

9.0 FUNDAMENTAL NUCLEAR MATERIAL CONTROL PLAN (FNMCP)

Combustion Engineering's FNMCP dated February 1980 was submitted June 11, 1980 and should be considered part of this license.

PART II. SAFETY DEMONSTRATION

1.0 OVERVIEW OF OPERATIONS

1.1 Corporate Information and Financial Qualifications

1.1.1 Name and Address of Licensee:

COMBUSTION ENGINEERING, INC.

1000 PROSPECT HILL ROAD

WINDSOR, CT 06095

Combustion Engineering is incorporated in the State of Delaware. The principal office location is in Windsor, Connecticut.

1.1.2 Name, Addresses and Citizenship of Principal Officers

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.

1.1.2 Company Background

Combustion Engineering is a diversified company serving electric utility companies, oil and gas producers, chemical companies and general industry throughout the world. The major portion of C-E's business has long been steam generation equipment for electric utilities, and it is one of the largest manufacturers of such equipment in the world. In recent years the company has diversified into related fields while continuing to apply its basic skills and technology.

C-E was first organized as a corporation in 1912. When considering the companies which have merged into the corporate structure, however, C-E's history dates back to the 1880's. Thus, the organization, as it exists today, has more than 90 years of experience in the design, development and fabrication of steam generation equipment.

C-E has been active in the development of nuclear power for more than 30 years. The Company's decision to extend its systems to large nuclear utility power plants represents a logical development of its previous activities as a supplier of thermal steam generating plants. All nuclear activities are carried out by the Nuclear Power Businesses Division.

The capabilities of the entire C-E organization are available to the Nuclear Power Businesses Division and will be utilized by it, as necessary, to fulfill its responsibilities.

Nuclear Power Businesses employs more than 1500 people of whom approximately 70% are scientists and engineers. A majority of the professional staff have at least five years experience in the nuclear field and have continued their education beyond the Bachelors Degree level. This staffing provides competence in the field of nuclear science and technology and extensive experience in the following specific areas: theoretical and experimental physics, mathematics, reactor analysis, chemistry, metallurgy, instrumentation controls, mechanical design, thermal sciences and nuclear and radiological safety.

The Nuclear Power Businesses Division is divided into three principal business segments; Nuclear Systems, Nuclear Services and Nuclear Fuel. Each of these business segments is headed by a Vice President responsible for the activities of the unit and who reports to the President, Nuclear Power Businesses.

The Nuclear Fuel Manufacturing activities within the scope of this license application are conducted under the auspices of the Vice President, Nuclear Fuel.

Nuclear Fuel Manufacturing

Nuclear Fuel Manufacturing (NFM) is equipped to provide a variety of services necessary to the development and manufacture of precision reactor components such as fuel rods and assemblies containing low enriched UO2 and control rods.

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1.1.4 Information Known to Applicant Regarding Foreign Control

There is no information known to Combustion Engineering, Inc. of any control exercised over it by any alien, foreign corporation, or foreign government. The stock of Combustion Engineering is traded on the New York Stock Exchange. According to the stock records of Combustion Engineering maintained by its Transfer Agent, the Chase Manhattan Bank, as of December 31, 1979 there were approximately 26,742 stockholders of record, holding 16,337,119 shares of Combustion capital stock issued and outstanding. Of this number less than 1 percent of all stockholders gave foreign addresses.

1.1.5 Financial Qualifications

Combustion Engineering's 10-K which details its financial position is attached as Appendix B.

1.2 Operating Objective and Process - Summary

The process at the nuclear fuel manufacturing facility begins with receipt of UO2 powder or fuel pellets enriched to a maximum of 5.0wt% U235. The primary source of this material is Combustion Engineering's.

oxide conversion plant in Hematite, Missouri. This powder is then mac up into batches with various additives and pressed into pellets. The pellets are dewaxed in a furnace where volatile additives are removed. The pellets then pass through a sintering furnace where they densify a attain the desired characteristics. Final sizing is accomplished through the use of a centerless grinder. The finished pellets are the loaded into zirconium tubes which are sealed and combined into finishe PWR fuel assemblies. The assemblies are finally loaded into approved shipping containers and delivered to a carrier for transport to their final destination.

1.3 Site Description

1.3.1 Population

The area surrounding Combustion Engineering's 1200-acre site is sparsely populated. Windsor, Connecticut is the nearest town of significant size, approximately five miles away, with a population of 22,502 and a population density of 760.2 per square mile. East Granby, Connecticut is the nearest town to the site, approximately three miles away, with a population of 3,532 and a population density of 198.4 per square mile. The distribution of population in the area is shown in Table 1.3. Figure 1.3.1 is a map of the general area showing the location of the towns listed in Table 1.3

1.3.2 Industry

The chief occupation of the populace in the immediate vicinity is farming. Farming is prevalent not only in this vicinity, b throughout the Connecticut Valley. Large industrial plants ar widely dispersed in the area.

1.3.3 Climatology and Meteorology

Climatological data for the area is based mainly on measurement made at the U.S. Weather Bureau Station located at Bradley International Airport, about five miles northeast from the site The following information was taken from the U.S. Weather Burea Local Climatological Data for the Hartford, Connecticut area:

> "The most significant feature of Hartford's climate is its rapid changeability. Weather is seldom average or normal for any appreciable length of time".

The mean temperature for the 50 year period, 1931 through 1979 was 50.0 degrees Farenheit, as recorded at Bradley International Airport. The maximum and minimum monthly mean temperatures were 83.2 degrees Farenheit and 19.1 degrees Farenheit respectively. The total precipitation for 1979 was 38.43 inches with a maximum of 5.15 inches falling in February. The mean annual precipitation from 1931 to 1979 was 42.34 inches. The maximum monthly precipitation was 21.87 inches in August 1955. The flood level for the area's worst flood (August 1955) was about 110 feet above mean sea level. Since the Combustion site is located approximately 180 feet above mean sea level, the probability of direct damage resulting from a local flood is very low.

The average hourly wind speed for 1979 was 8.8 miles per hour The highest recorded velocity was 70 miles per hour in November 1950. The prevailing wind direction for six months, May to October is South, and for the six months November to April, is Northwest. The average wind velocity at the Combustion site is 11.2 miles per hour.

With low-to-moderate wind speeds, inversion conditions may exist from sunrise to sunset. A strong lapse exists around noon; the temperature difference is maximum with air flow upward at a maximum rate. As night approaches, weak lapse condition occur with low air flow.

Hartford's location relative to the continent and the ocean has a significant influence on the area's meteorological and climatological conditions. With the prevailing west-to-east air flow, continental modifications of the air are important. However, sudden and oftentimes serious upsets result when storms move north or when other pressure developments produce the strong and persistent northeast winds associated with storms known locally as "coastals" or "northeasters".

Seasonable air mass characteristics vary from the extremely cold and dry continental polar quality of winter to the warm, humid maritime tropical characteristics of summer -- the one type from Canada and the other from the Gulf of Mexico, the Caribbean Sea, or the Atlantic Ocean.

Local topography also influences the climate. The Berkshire

Hills to the west and northwest are a source of summer thunderstorms which, when accompanied by wind and hail, sometimes do condsiderable damage to the crops in the Connecticut Valley. Frequently during the winter, when rain falls through the cold air trapped in the Valley, the resultant icing creates hazardous conditions for transportation and utility installations. On clear nights in the late summer or early autumn, cool air drainage into the Valley, plus Connecticut River moisture, produce ground fog which sometimes becomes quite dense through the Valley and hampers ground and air transportation.

1.3.4 Geology

The surrounding area has been subjected to the actions of glacial ice. All dominant geological features are a result c erosion and depositions caused during the Pleistocene era. The State of Connecticut has favorable earthquake history. Ten earthquakes are listed, the first recorded in 1791 and t last in 1925. All of these, with the exception of the first were local in nature and of moderate intensity.

1.3.5 Hydrology

The surface drainage in the surrounding area is excellent. The predominantly sandy nature of the soil and heavy forest cover results in very moderate run-off even after heavy prolonged precipitation.

The site creek, into which all site effluents are discharged flows into the Farmington River which flows along the northwest corner of the Combustion site, shown in Figure 1.3.2. Two and one-half miles below the site, the river flows over the dam of the Farmington River Power Company and approximately six miles below that, into the Connecticut River. The minimum recorded flow in the river is 5.1 cubic feet per second.

Public use of the Farmington River is limited to fishing and boating. Analysis of 27 water samples taken between January 1966 and March 1967 at the Windsor Bridge by the State Department of Health indicated a bacteria count which ranged from 410 to 11,000 per 100 milliliters in a given microscopic field for a total mean average of 409.4 which renders it unsuitable for potable use or bathing.

1.3.6 Topography

Elevations in the region vary from about 100 feet above sea level at the Farmington River to hilltop heights of from 200 to 280 feet, with the level areas being at an elevation averaging approximately 120 feet above sea level. The land rises to the west of the site to a north-south ridge approximately two miles distant and averages 450 feet in elevation. The land slopes gently away from the site on all sides as evidenced by the direction of the flow of small streams which drain the region. The Farmington River flows along the northern boundary. The land on all sides consists of mostly open level fields tilled for farming except for a small extension of wooded area on the northeast.

1.4 Locations of Buildings Onsite

The locations of all buildings on the Windsor Site are shown in Figure 1.3.3.

1.5 History of License

Combustion Engineering first applied for a license to process low enriched uranium by the methods described in section 1.2 in 1968. License SNM-1067 was then issued for a period of 5 years by the U.S. Atomic Energy Commission (AEC). The License was renewed at 5 year intervals with the latest renewal approved in March 1983 for an additional 5 year period.

TABLE 1.3

POPULATION AND POPULATION DENSITY OF TOWNS WITHIN 10 MILE

RADIUS OF COMBUSTION ENGINEERING, INC.

Town	Approximate Distance of Town Center to Site	General Direction from Site	Square Miles	Estimated Population (1970)	Population Density Person/sq mi
Hartford	9	S	18.6	158,017	8495.6
Windsor	5	SE	29.6	22,502	760.2
Bloomfield	5	S	26.9	18,301	680.3
West Hartford	8	SW	21.6	68,031	3149.1
East Hartford	9	SE	18.2	57,583	3164.0
Manchester	13	S	27.6	47,994	1738.9
South Windsor	7	SE	29.2	15,553	532.6
East Windsor	6	SE	25.6	8,513	319.6
Windsor Locks	4	E	9.6	15,080	1520.8
East Granby	3	N	17.8	3,532	198.4
Simsbury	6	SW	34.2	17,425	510.9
1/3 Avon incl. Center	8	SW	7.4	3,000	405.4
Granby Center	6	N	10.0	4,500	450.0
1/2 Suffield incl. Center & W. Suffield	6	N	22.0	4,300	195.4
1/3 Enfield incl Thompsonville	8	NE	11.0	16,000	1454.5
1/2 Ellington excl Center	12	E	17.0	3,900	229.4
1/3 Vernon excl Center	13	E	6.0	7,200	1200.0
TOTALS			333.3	471,481	

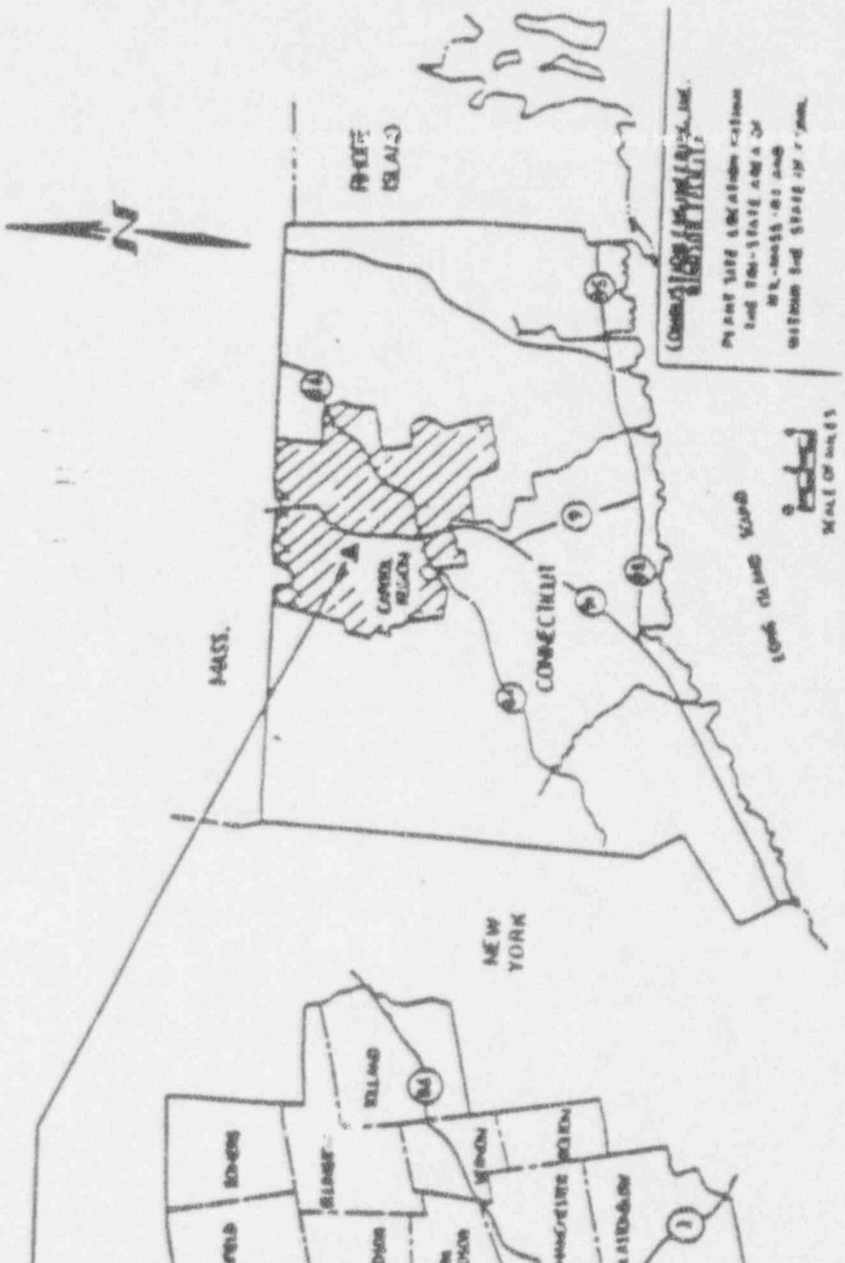


Figure 1.3.1

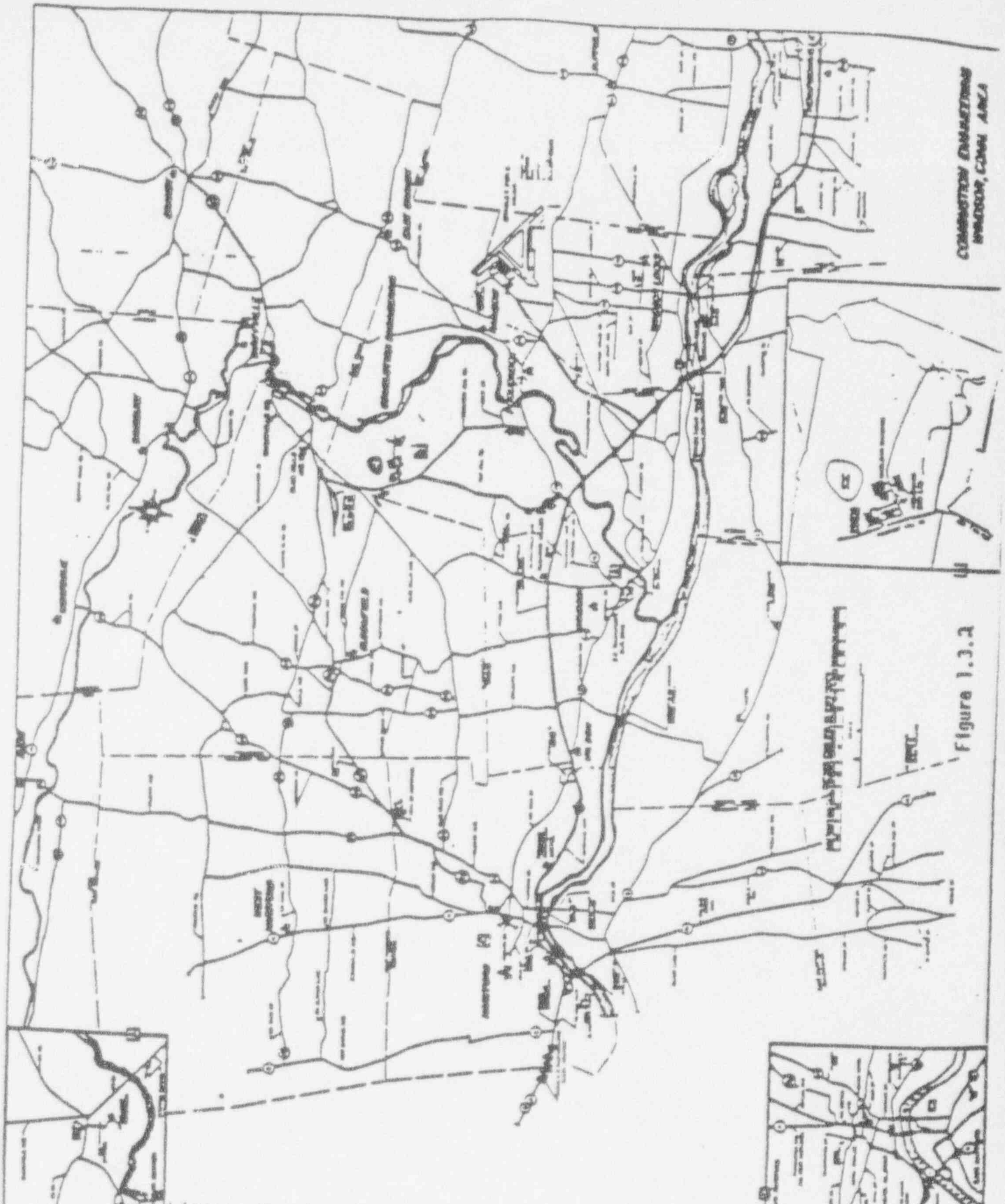
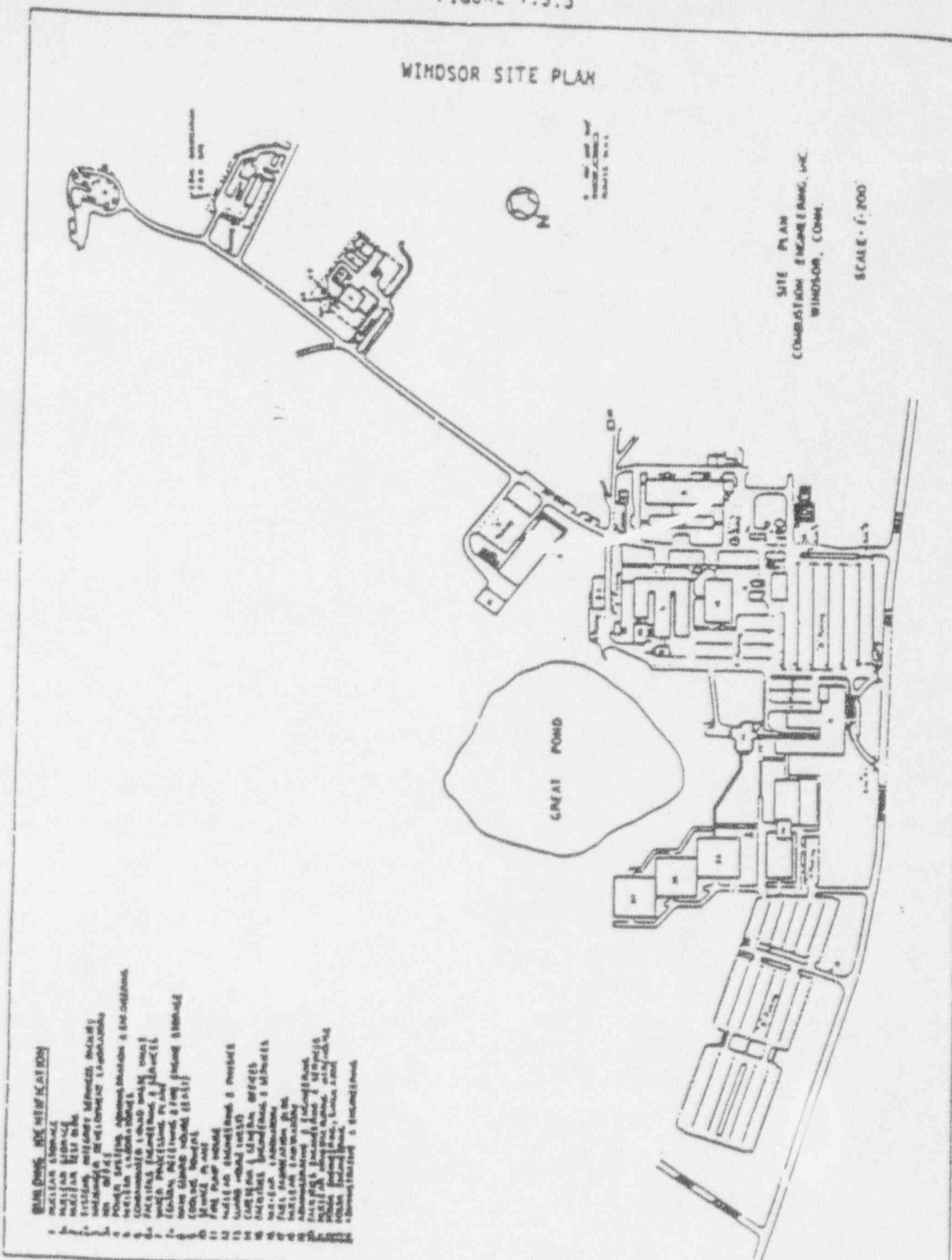


Figure 1.3.3

FIGURE 1.3.3

WINDSOR SITE PLAN



2.0 FACILITY DESCRIPTION

2.1 Plant Layout

The principal activities of the nuclear fuel manufacturing operation are carried out in Building #17, a one level building measuring 120 ft x 340 ft. (40,800 square feet). The shop section contains approximately 36,000 square feet (120 ft. x 300 ft.), has concrete flooring, corrugated asbestos siding, and a poured gypsum roof deck approximately 25 ft. above floor level.

The front office section of the facility contains approximately 4800 square feet (120 ft. x 40 ft.), has concrete flooring, exterior concrete block with full windows, and a poured gypsum roof deck approximately 11 feet high.

A warehouse, Building #21, is provided for the storage of incoming fuel shipping containers, raw materials, and finished components. This building is a prefabricated rigid frame steel structure approximately 120 feet long and 80 feet wide with a height of 16 feet at the eaves. It is located approximately 100 feet west of Building #17.

Details of the manufacturing facility layout are shown in Figure 8-1 of this application.

2.2 Utilities

Electrical power to the Windsor site is provided by the Northeast Utilities substation at North Bloomfield and the step-down transformer at Building #17, substation #6. The substation transformer (3,750 KVA) steps down from 22.9 KV, to 480 volt, 3-phase, 3-wire. The output is then connected to the low voltage metal clad switch gear for distribution through the building. A further step-down to 208/120 volt is made for lighting and general convenience power.

Emergency Power System

A diesel generator serves as a back-up emergency power system for the manufacturing facilities. The generator produces 3-phase, 480 volts, and 200 KW. The described generator feeds a rated distribution panelboard which has several 100 ampere, 3-phase 480 volt circuit breakers. The panelboard is switched from normal power to generator (emergency) power by an Asco transfer switch. Diesel start-up and transfer takes approximately ten to twelve seconds. A circuit breaker within this panelboard is used to supply emergency power to the manufacturing facilities.

The principal site water supply is provided by the Metropolitan District, the source of city water for the greater Hartford area. Chemical and radiological analyses for both raw and treated well water have been made, and any changes in composition or activity from any cause will be discovered rapidly.

2.3 Heating, Ventilation, and Air Conditioning (HVAC)

The Building #17 office area, consisting of 4800 square feet is heated and cooled by hot and chilled water respectively supplied by the Windsor site central boiler house. Office areas have built-in convectors which heat and cool the areas depending on the time of year. Each office has an exhaust system which ventilates the area and allows fresh air to be brought in.

The main shop area of Building #17 consists of approximately 36,000 square feet and is divided into two main areas. The first area is the unclad fuel area (pellet shop) which contains 12,000 square feet, the second area is the manufacturing area, which contains 24,000 square feet. The unclad fuel area is air conditioned on a year round basis because of process heat generated by the continuously running sintering furnaces. Ventilation of the area is continuous with air being exhausted through banks of pre and absolute filters. The entire area (pellet shop) is maintained at a negative pressure of approximately 0.1" H_2O by the action of the exhaust systems. The manufacturing area is not air conditioned, except for the bundle assembly room. The air outside the bundle assembly room is heated by ceiling blowers which obtain hot water from the main power house and circulate it through convectors. During warm weather, the windows are left open for natural air circulation. The bundle assembly room has a rooftop HVAC unit. Ventilation is accomplished by bringing outside air into the HVAC unit where it is circulated and then passed into the room.

2.4 Waste Disposal

All non-radioactive wastes are disposed of in accordance with applicable state and local regulations. The radioactive waste is then delivered to a licensed facility for ultimate disposal in compliance with all regulations.

2.5 Chemical Storage

All chemicals will be stored in accordance with federal and state regulations, including OSHA.

2.6 Security

The entire Windsor Site is guarded 24 hours a day by the site security force. Additional security provided for the manufacturing facility is described in Combustion Engineering's Security Plan dated October, 1980 and submitted under separate cover and should be considered part of this renewal application.

2.7 Fire Protection

A full time Fire Marshal is on site during normal working hours. He is on 24-hour alert. His duties include periodic inspection of the buildings and routine checks of all fire fighting equipment.

Fire protection, including sprinklers, is designed into all buildings which are subject to fire damage.

A direct emergency telephone line to the Windsor Fire and Safety complex (Fire, Police, Ambulance, etc.) is controlled by site security personnel. A copy of the Certificate of Insurability from American Nuclear Insurers is provided in Figure 2.7.

The manufacturing facilities are constructed and operated consistent with requirements of the applicable NFPA fire safety codes.

FIGURE 2.7

Nuclear Energy Liability - Property Insurance Association
Property Division

The Exchange, Suite 245, 270 Farmington Avenue, Farmington, Connecticut 06032

Declarations attached to and made a part of Policy No. 1474

Rate See Rate Computation Endorsement No. 1 Premium $337,010.

Name of Insured COMBUSTION ENGINEERING, INC.

Mailing Address 900 Long Ridge Road, Stamford, Connecticut

Unless otherwise provided herein, loss, if any, shall be adjusted with and payable to the named Insured.

The policy period shall be for the term of one year from July 1, 19 80 to July 1, 19 81, beginning and ending at noon, Standard Time at the location of property covered as specified herein.

Description and location of property covered.

Location No. 1 Amount of Insurance $209,000,000. Deductible $25,000.

All Real and Personal Property on the Insured's Plant premises known as Buildings 1, 2,5,6,17,18 and 21 and all Intervening Roadways connecting these Buildings and all Parking Areas adjacent to them as shown by the heavy black line on Combustion Engineering Drawing FP-3, Fire Protection System, dated September 12, 1972 and located in Windsor, Connecticut and including the extension to Building No. 1, designated as Building 1A, which is outside the heavy black line (PROPERTY FILE NO. N-6).

Location No. 2 Amount of Insurance $90,000,000. Deductible $25,000.

All Real and Personal Property on plant premises occupied principally for the processing of uranium hexafluoride, located on Missouri State Highway No. 21-A about one-half mile east of Hematite and six miles west of Festus in Missouri including the 1.47 acre Parcel of land (but excluding any dwellings situated thereon) described by the Survey dated May 30, 1979, Order No. 1346, and tract in United States Survey 423, Township 40 North, Range 5 East, Jefferson County, Missouri (PROPERTY FILE NO. N-1).

Countersignature

Countersigned July 1, 19 80, at Farmington, Connecticut

by B.C. PROOM, President

Authorized Representative

BY [signature]

Its Attorney

NELPIA 38-A Rev. 8/1/77 (NIRB)

3.0 Organization and Personnel

Functions of key positions important to safety, specifics on education and experience required for key positions important to safety, operating procedures, and training are described in Part I, Chapter 2.0 - Organization and Administration. The Windsor Nuclear Fuel Manufacturing Facility and Product Development Organization structure is depicted in Figure 3.1.1.

3.1 Functions of Key Personnel

The function, responsibilities and authorities of key personnel important to safety are described in Part I, Section 2.1 of this application. This section provides similar information for the remaining personnel holding key line management positions.

3.1.1 Manager, Nuclear Materials Licensing

The Manager, Nuclear Materials Licensing reports to the Vice President, Regulatory Affairs and has responsibility for licensing of Combustion Engineering's Nuclear Fuel Manufacturing and Product Development activities. This responsibility is executed by identifying applicable NRC regulations and ensuring that they are appropriately addressed in applicable licenses and certificates of compliance, as necessary.

3.1.2 Manager, Nuclear Materials

The Manager of Nuclear Materials reports to the Controller, who in turn reports to the President, Nuclear Fuel. Nuclear Materials control relating to the receipt, storage, use and transfer of Special Nuclear Material (SNM); the accounting and locating of SNM; preparation/revision/submittal of the Fundamental Nuclear Material Control Plan; quantity accountability and maintenance of records relating to the operating, receipt and storage of SNM are directed by the Nuclear Materials Manager. In order to execute these functions, he defines the Materials Control and Accountability Program used by the Windsor Nuclear Fuel Manufacturing Facility.

The Manager of Nuclear Materials has no production responsibility and he has no hands on responsibility for nuclear materials. He or she also provides an audit function for Combustion Engineering's nuclear fuel manufacturing facilities to ensure compliance of operations personnel with the requirements of the Materials Control and Accountability Program.

3.1.3 Director, Quality Assurance

The Director of Quality Assurance reports to the President, Nuclear Fuel. Quality control and quality assurance functions are under the direction of the Quality Assurance Director. He or she is responsible for establishing quality control inspection procedures to ensure that manufacturing operations produce a product that meets or exceeds customer specifications. He or she also prepares an implements the Quality Assurance Manual for the Windsor Nuclear Fuel Manufacturing Facility.

The Director, Quality Assurance, has no production responsibility. He has the authority to shutdown operations which inspection reveals are not producing a product consistent with customer qualifications.

3.1.4 Manager, Accountability and Security

The Manager of Accountability and Security reports to the Uranium Plant Manager. The implementation of the Fundamental Nuclear material Control Plan, maintaining custodial control of nuclear materials, and management of radioactive waste are the responsibility of the Manager of Accountability and Security. He or she maintains nuclear materials measurement control systems and records of nuclear materials in the production process. He or she is also responsible for the preparation and implementation of the Physical Security Plan and oversight of the security force for the Windsor Nuclear Fuel Manufacturing Facility.

3.1.5 Operations Shift Supervisors

The Operations Shift Supervisors report to the Uranium Plant Manager. They are responsible for the coordination of activities amongst Line Managers to ensure that the facility production goals are satisfied within the limits imposed by Federal, State and local regulations, this license application, certificates of compliance and other permits, as applicable.

3.1.6 Manager, Training

The Manager of Training reports to the Uranium Plant Manager. He or she is responsible for the training program for facility personnel as well as other Combustion Engineering employees or visitors that require unescorted access to the facility. The Manager assures that appropriate training materials are prepared and administered, testing is performed and records are retained as called for in Part I of this license application.

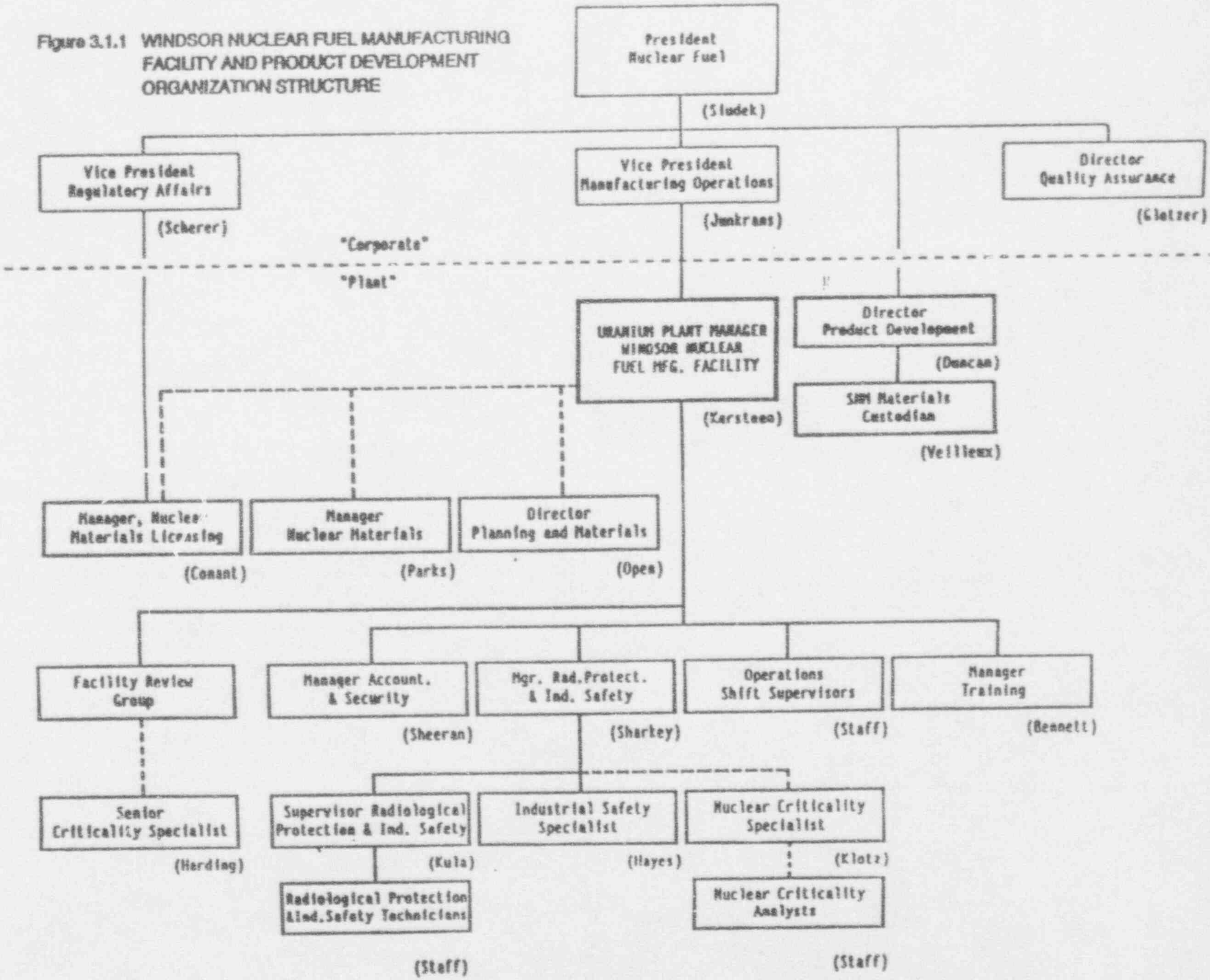
3.1.7 Director, Planning and Materials

The Director, Planning and Materials reports to the Vice President, Manufacturing Operations. He or she is responsible for scheduling of production work, planning materials requirements, and purchasing equipment and materials needed to support the manufacture of fuel assemblies and other products. The Director also develops and integrates into the manufacturing process modernized manufacturing information systems designed to improve efficiency and responsiveness to customer needs.

3.2 Resumes of Key Personnel Important to Safety

Resumes of key personnel important to safety are provided on Pages II.3-5 through II.3-30.

Figure 3.1.1 WINDSOR NUCLEAR FUEL MANUFACTURING FACILITY AND PRODUCT DEVELOPMENT ORGANIZATION STRUCTURE



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MAURICE E. HATCHER - Radiological Protection and Industrial Safety Technician

EDUCATION

Jonathan Law High School, Milford, CT, 1981

EXPERIENCE

COMBUSTION ENGINEERING, INC. 1988 to Present
Windsor Nuclear Fuel Manufacturing

Radiological Protection and Industrial Safety Technician

Power Systems Energy Services, Inc. Sept. 1987 to 1988

Senior Health Physics Technician

Senior Health Physics Technician assigned to support various utilities on a contract basis. Responsibilities have included identifying radioactive waste, controlling high level waste, performing routine radiation, contamination and airborne surveys and releasing equipment.

Bartlett Nuclear, Inc. Feb. 1987 to Sept. 1987

Health Physics Technician

Health Physics Technician assigned to support various utilities on a contract basis. Responsibilities have included performing routine radiation, contamination and airborne surveys. Worked in the ALARA unit responsible for pre- and post-job survey data; installation of numerous contamination control devices; installation of various HEPA ventilation units to control the spread of loose surface contamination. Performed surveys of tools and equipment to allow for free release. Performed decontamination of articles above established limits. Issued respiratory protection equipment.

USS HUNLEY (Submarine Tender) May 1983 to Oct. 1986
Article 108 Qualified

Radiological Controls Monitor

Duties included: ensuring proper radiological controls were enforced during all radioactive work, including supervision of Control Point Watches; ensuring numerous radioactive material transfers were properly executed; maintaining nuclear grade "A" cleanliness of primary systems; monitoring personnel exiting highly contaminated work areas; monitoring personnel exposure; ensuring that all personnel received exposure within established limits. Performed job coverage of two Steam Generator primary side inspections and various Radioactive Liquid Waste tank dives. Assigned primary collateral duties as Radioactive Materials Control Petty Officer and Primary Radiological Controls Monitor for the Northern Europe Radiological Casualty Assistance Fly-Away Team.

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.

RICHARD S. KULA - Supervisor, Radiological Protection and Industrial Safety

EDUCATION

University of Connecticut
B.S. - Animal Sciences, 1977
Postgraduate Study, Auburn University
Major - Animal Science; Minor - Business

EXPERIENCE

COMBUSTION ENGINEERING, INC.

Supervisor, Radiological Protection — July 1991 - Present

Responsible for supervision of the Windsor Nuclear Fuel Manufacturing and Product Development radiological protection program. Responsibilities include the training and supervision of the Radiological Protection Technicians assigned to NFM. Responsible for direct surveillance of activities related to radiological criticality and industrial safety, environmental protection and emergency planning.

Senior Radiation Protection Technician — 1989 - July 1991

Initially assigned the third shift Lead Technician position, overseeing three technicians having the responsibility of controlling the use and handling of nuclear fuel through its processing from uranium oxide powder to a pelletized form, and subsequent fabrication into commercial reactor fuel assemblies. Later assigned the Lead Technician for the decommissioning of the oxide handling and pelletizing equipment, and the redeployment of the facility for return to processing of pre-pelletized fuel into reactor fuel assemblies.

INTERSTATE NUCLEAR SERVICES

Assistant Plant Manager — 1987 - 1989

Responsible for all second shift plant operations at this nuclear decontamination laundry, servicing power plants, duties for two Health Physics Technicians and serve as Radiation Safety Officer for the second shift. Oversee the receiving and shipping of radioactive materials and the generation of documents required by state and U.S. D.O.T. for the shipping of these materials. This facility employed 65 persons on two shifts.

RICHARD S. KULA (continued)

COMBUSTION ENGINEERING, INC.

<u>Health Physics Technician</u> 1981 - 1987

Responsibility included performing a continuous program of air sampling and radiation monitoring to access and control contamination and radiation both in the plant and in the environment; controlling the use, handling, and storage of all radioactive materials. Oversee the sorting, packaging and shipping of radioactive waste. Evaluating and reporting radiation exposures and contamination levels. Inspecting manufacturing facilities to ensure compliance with applicable license and government regulations relative to nuclear and industrial safety.

AUBURN UNIVERSITY

<u>Assistant Laboratory Manager</u> 1978 - 1980

Responsible for the day-to-day laboratory operations involved with muscle physiology and muscle microanatomy research and teaching. Supervised all aspects of laboratory maintenance, data management, and maintenance of grant records. Assisted in teaching laboratory sessions for introductory and advanced animal science courses.

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ROBERT S. HARDING - Senior Criticality Specialist

EDUCATION

B.S., Physics, Trinity College, 1951
Ph.D., Physics, University of Rochester, 1958

EXPERIENCE

COMBUSTION ENGINEERING, INC.

Principal Consultant, Nuclear Engineering 1989 to Present

Currently assisting Nuclear Fuel Manufacturing in updating and implementing applicable SNM licenses. Serve as second party independent reviewer for facility criticality evaluations. Also act as criticality consultant to the Nuclear Safety Committee.

Senior Nuclear Scientist, Nuclear Engineering 1985

Acted as R&D Manager in the Nuclear Fuels area. In addition, had technical cognizance for the criticality, shielding, and vessel fluence activities in Nuclear Engineering. Also served as criticality consultant to the Nuclear Safety Committee.

Manager, Nuclear Design, Nuclear Engineering 1972 to 1985

Responsible for the administrative and technical management of the Nuclear Design Department for design activities on PWR Nuclear Steam Supply Systems - fuel management, reactivity control, power distribution monitoring, generation of core nuclear safety parameters, radiation physics, criticality evaluations for excore fuel handling operations, nuclear design data for NSSS proposals, and support of other functional groups for plant design. Also served as criticality consultant to the Nuclear Safety Committee in 1983 and 1984.

Manager, PWR Physics Methods Development, 1968 to 1972
Physics and Computer Analysis Department

Responsible for developing and testing of design methods and computer codes employed in the nuclear design of PWR reactor cores.

Manager, HWOCR Reactor Physics Section, 1965 to 1968
Physics Department

Responsible for the nuclear design and methods development activities on the Heavy Water Organic Cooled Reactor project and served as principal liason between the Physics Department and the HWOCR Project Office. Also responsible for the nuclear design analyses relating to the conversion of a Savannah River Reactor to a D_2O moderated power reactor.

Staff Physicist, General Nuclear Engineering, 1962 to 1965
(C-E Subsidiary)

Participated in the State-of-the-Art Physics Program, a study of reactivity control methods for the NASA reference reactor design, and the review of literature on critical experiments for the Technical Progress Review, Power Reactor Technology.

Supervisor, Advanced Critical Experiment Facility, 1961 to 1962
Nuclear Power Department

Responsible for the technical programs and safe operation of both reactor cells.

Staff Physicist, Advanced Critical Experiment Facility, Nuclear Power Department 1958 to 1961

Participated in the SIC Flux Measurement, Nuclear Superheat, BONUS Critical, and Army Boiling Water Reactor (PL-2) Critical Programs. On BONUS Program was the Project Principal Experimental Physicist. Also served as a member of the Critical Facilities Safeguards Committee and in 1961 was designated as the Alternate Supervisor of the Critical Facilities for purposes of assuring safe operation of the facilities.

PAUL F. O'DONNELL - Nuclear Criticality Analyst

EDUCATION

Professional Engineer Degree, Nuclear Engineering, North Carolina State University — 1978

B.S., Nuclear Engineering, University of Lowell — 1977

EXPERIENCE

COMBUSTION ENGINEERING, INC.

Consulting Nuclear Engineer — 1988 to Present

Responsible for performing criticality analyses for both the Hematite and Windsor fuel manufacturing facilities. These analyses include the criticality evaluations in support of the design of the new pelletizing operation at the Hematite facility. Developed KENO-IV models for evaluation of process, storage and transportation systems. Employed both the 16 group Hanson and Roach library and the NITAWL and XSDRNPM generated cross section libraries for these analyses. Also has evaluates and reviewes equipment and process changes at both facilities.

Additionally, responsible for the criticality evaluation of spent PWR fuel assembly storage facilities. In these analyses, employed the two dimensional transport theory code DOT.

Principal Nuclear Engineer — 1985 to 1988

Directly responsible for the development, testing and verification of core physics models for nuclear power reactors. In addition, directly involved in the analyses of advanced fuel management strategies for pressurized water reactors. In these capacities, employed the Discrete Integral Transport theory code DIT for the generation of neutron cross sections in HARMONY format, the three dimensional diffusion theory code ROCS for the calculation of core power distribution-and reactivity, the two dimensional tranpsort theory code DOT for the calculation of radial boundary conditions, the one-dimensional transport theory code ANISN for axial flux and boundary conditions calculations, and PDQ for pin power distribution calculations.

Participated in a project which evaluated core power distribution and reactivity following a steam line break accident. In this analysis, developed three dimensional thermal-hydraulic and neutronic models for the HERMI-- code.

PAUL F. O'DONNELL

Senior Nuclear Engineer 1982 to 1985

Directly involved in a DOE sponsored program for the development of gadolinia shims in pressurized water reactors. This included the definition of assembly designs with gadolina shimmed pins, generation of neutron cross sections with the DIT code, and the development of an 18 month fuel management strategy.

Also developed a computer code for the automated generation of assembly neutron cross sections in HARMONY format. In addition, employed and modified the point kernel code SHADRAC for neutron attentuation calculations.

Nuclear Engineer II 1980 to 1982

Principal investigator for the development of fuel management strategies that would transition the CANDU-600 heavy water reactor from the natural uranium fuel cycle to higher enrichment and burnup fuel cycles . This program was sponsored by the U.S. Arms Control and Disarmament Agency. In this capcity, responsible for the development of cross sections using the DIT code and the definition of acceptable refueling strategies using the three dimensicanl diffusion theory code FC.

GENERAL ATOMIC COMPANY, HTGR Core Performance and Analysis Department

Physicist 1978 to 1980

Involved in High Temperature Gas Cooled reactor (HTGR) conceptual design. In this capacity, responsible for: prismatic and pebble bed reactor core design and safety analyses, estimations of heavy metal mass requirements, cross section generation for hexagonal graphite fuel assemblies and the spherical shaped graphite lattice used in the Pebble Bed concept. In these analyses, employed the tranport theory code MICROBURN for the generation of neutron cross sections, the one dimensional tranport theory code DTF for control rod modeling, and the two dimensional diffusion theory code 2DB for core power distribution and reactivity calculations.

KEVIN R. HAYES - Industrial Safety Specialist

EDUCATION

B.S., Industrial Technology, Central Connecticut State University, 1988
Special Curricula: Manufacturing; also completed several requiremnts for Occupational Safety and Health

A.S., Manufacturing Engineering, Hartford State Technical College, 1986

EXPERIENCE

COMBUSTION ENGINEERING, INC.

Industrial Safety Specialist, Windsor Nuclear Fuel Manufacturing Facility — 1989 to Present

Consultant on industrial safety and environmental protection. Responsible for implementation of programs and standards in the industrial safety and environmental protection area. Also advise the Radiation Protection and Industrial Safety Technicians in the proper methods of monitoring industrial safety and environmental protection compliance.

Safety Specialist, Power Systems Energy Services — 1987 to 1989

Responsible for coordinating Combustion Engineering Site environmental programs for compliance with local, state, and federal regulations. Assisted in revising C-E Nuclear Fuel Manufacturing facility emergency plans and procedures. Also served as security guard and ambulance technician. In this capacity was responsible for site security, responding to emergencies and writing/maintaining incident records.

SIMSBURY, CT FIRE DISTRICT

Fire Department Dispatcher — 1986 to 1987

Dispatched assistance for all fire calls, interfaced with public and wrote/maintained records.

METAL IMPROVEMENT COMPANY — 1982 to 1986

Employed in several shop manufacturing positions including Leadman, Tool Grinder, Maintenance Technician, and Machine Operator.

KEVIN R. HAYES

Supplemental Experience

Volunteer firefighter for 10 years; Connecticut certified Emergency Medical Technician (certified EMT-D); Senior Rescue Instructor, Hartford County Fire/Emergency Plan.

GARY C. KERSTEEN - Uranium Plant Manager, Windsor Nuclear Fuel Manufacturing, and Emergency Director

EDUCATION

B.S. Mechanical Engineering, Trinity College, Hartford, CT, 1968

Supplemental Education:

Advanced Course - Nuclear Materials Safeguards, Argonne Labs
Effective Middle Management Courses
Financial Management Course
Crosby Quality Education Instructors Course,
Taught Quality Improvement Process to more than 100 employees
Worcester Polytechnic Institute, Statistical Process Control
Motorola Management Institute, Motorola University

EXPERIENCE

COMBUSTION ENGINEERING, INC.

Director, Planning and Materials 1990 to 1992

Directed the master planning of all nuclear fuel manufacturing activities. Managed the planning and procurement of contract materials and other supplies and services. Coordinated uranium management activities. Directed manufacturing related information systems development and support including installation of Statistical Process Control (SPC) and Material Requirements Planning (MRP) systems.

Production Manager 1982 to 1990

Managed all aspects of production control, material control and the manufacturing work force and the Windsor Plant. During this time we installed real-time fuel rod data information systems, moved pellet operations to the Hematite, Mo. Plant and initiated the fuel rod automation project. Developed a sophisticated fabrication planning system. Initiated the CE Quality Improvement Process and developed Improvement Teams at the Windsor Plant to encourage employee empowerment, involvement and communications.

GARY C. KERSTEEN

Production Control Manager 1979 to 1982

Managed the production control section, the material control group and the warehouse activities.

Supervisor, SNM Accountability 1975 to 1979

Started the Windsor Plant accountability department. Wrote the first Fundamental Nuclear Material Control Plan. Brought the first distributed data processing system to the Windsor Plant to automate accountability of special nuclear material. Developed the initial limit of error methodologies used at Windsor.

Manufacturing Engineer 1974 to 1975

Initial assignment at Nuclear Fuel Manufacturing developing fixtures and processes for fabrication activities.

MILITARY EXPERIENCE

U.S. ARMY 1969 to 1974

U.S. Army Officer Candidate School- Engineering, Fort Belvoir, VA

Commander, 575 Ordnance Company (Guided Missile Repair), Germany

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ROBERT W. SHARKEY - Manager, Radiological Protection and Industrial Safety

EDUCATION

University of Lowell
M.S. - Radiological Science and Protection 1990
B.S. - Radiological Health Physics 1988

LICENSE

U.S. Nuclear Regulatory Commission, Reactor Operator, License No. 10723.

EXPERIENCE

JACOBS ENGINEERING GROUP, INC.

Health Physicist 1989-1990

Developed the Weldon Spring Site internal dosimetry program. Developed worker health and safety plans for remediation activities. Developed air monitoring plan to comply with 40CFR61 radionuclide NESHAPS. Provided radiation safety training for all site personnel.

UNIVERSITY OF LOWELL

Nuclear Reactor Operator 1988-1989

Setup and conducted experiments using the ULR 1MW research reactor and a 800,000 Curie CO-60 gamma source. Maintenance of all electrical and mechanical facilities. Inspect, repair and calibrate nuclear instrumentation and radiation detection equipment. Training of undergraduate engineers in nuclear reactor operations.

Teaching Assistant 1987-1988

Instruction of the laboratory course, Nuclear Instrumentation.

E.I. DUPONT DE NEMOURS & COMPANY, NEW PRODUCTS, BILLERICA, MASSACHUSETTS

Radiochemistry Technologist 1987-1988

Utilization of radiation detection equipment and smear surveys to minimize exposure and contamination. Preparation of radiopharmaceuticals in a hot cell after proton bombardment. Radioassy of pharmaceuticals using nuclear instrumentation.

ROBERT W. SHARKEY

U.S. AIR FORCE

Avionic Navigation Systems Specialist 1980-1985

Test, troubleshoot and repair avionics to component level. One year special assignment as an aircraft maintenance controller directing all flight maintenance activities.

MEMBERSHIP

St. Louis Chapter, Health Physics Society

4.0 RADIATION PROTECTION PROCEDURES AND EQUIPMENT

4.1 PROCEDURES

A manual entitled "Nuclear Licensing & Safety Procedures" which contains procedures necessary to implement the radiation protection program described in Part I of the license application is maintained under the direction of the Manager, Radiological Protection and Industrial Safety.

4.2 Written Procedures

All routine operations involving nuclear fuel handling are covered by a shop traveler and/or various operation sheets (O.S.) which are issued by Manufacturing Engineering or Quality Assurance.

These procedures include the necessary precautions which must be observed to assure that the operation is conducted in a safe manner. The Manager, Radiological Protection and Industrial Safety will review these precautions regarding all aspects of safety and indicate his approval in writing. However, procedures involving a change in the criticality safety controls used for a particular process in the past shall be approved by Manager, Radiological Protection and Industrial Safety, and the Nuclear Criticality Specialist. Each Line Supervisor shall instruct his people to assure their understanding of the operations and their particular safety limits and restrictions.

It is the responsibility of the Supervisor, Radiological Protection and Industrial Safety to assure that each work station is properly posted, and that operations are performed in compliance with posted limits and written instructions.

4.3 Posting and Labeling

All work stations involving handling of special nuclear material are posted with a Nuclear Criticality Safety Limits in accordance with Section 4.0 of Part I of this application. All mass limited containers are labeled as to U235 contents and enrichment. Radiological posting of areas is in accordance with 10 CFR 20.203.

4.4 Personnel Monitoring

All personnel wash their hands before exiting the Pellet Shop and monitor their hands, exposed areas of the body and personal clothing with the alpha personnel monitor located at the change line. Any person having suspected contamination on his body must thoroughly wash the area and recheck for contamination. If contamination persists, a member of the Radiological Protection and Industrial Safety staff assists in further decontamination.

4.5 Surveys

Removable contamination levels in plant areas and on items to be released to an unrestricted area are established by smearing an area of 100 cm2 (4" x 4") with two inch diameter smear paper. Pellet Shop floor smears are taken on a weekly basis. Cold Shop floor smears are taken on a monthly basis. Contamination action thresholds are provided in Section 3.2.8 of Part I of this application.

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

4.6 Reports and Records

Radiological Protection records for the current calendar year, including training, and reports required by the regulations of the U.S. Nuclear Regulatory Commission and this license are retained under the direction of the Manager, Radiological Protection and Industrial Safety. Reports and records for previous years are made available to inspectors upon request.

4.7 Instruments

Types of radiation detection instruments, their capabilities, and frequency of calibration are described in Section 3.2.4 of Part I of this renewal application.

4.8 Protective Clothing

Protective Clothing requirements for personnel entering the Pellet Shop are described in Section 3.2.1 of Part I of this application.

4.9 Dosimetry

4.9.1 TLD Badges

Each individual who enters a restricted area under such circumstances that he is likely to receive a dose in any calendar quarter in excess of 2½ percent of the applicable value specified in

10 CFR 20.101 (a) shall be supplied with a TLD badge and indium foil for purposes of personnel dosimetry. Badges are processed monthly. When a high exposure is suspected, the individual's badge is sent out for immediate processing. All visitors are supplied with indium foil badges. Area TLD badges and neutron foils are also strategically placed throughout the facility for the purpose of recording background radiation levels as well as radiation resulting from a criticality accident. These badges are processed monthly for normal operations and immediately following a criticality accident. Procedures to determine high radiation doses immediately following a criticality accident are described in the Emergency Procedures Manual.

4.9.2 Breathing Zone Monitoring

Breathing zone monitoring of personnel is conducted in accordance with Section 3.2.5 of Part I of this application.

5.0 OCCUPATIONAL RADIATION EXPOSURES

Due to the extremely low levels of penetrating radiation which exist at Combustion Engineering's fuel fabrication facility (<5 mr/hr), the greatest emphasis in exposure control is directed towards minimizing ingestion of airborne uranium particulates. To this end, C-E strives to maintain internal exposures as low as reasonably achievable through the use of ventilated hoods and process containment and an extensive air sampling program. General air samplers are strategically placed throughout the facility to provide indications of airborne activity levels and are analyzed three times each working day. A bioassay program which includes periodic urinalysis and in-vivo counting also provides information regarding internal deposition of radioactive materials and confirms Combustion Engineering's long standing commitment to the ALARA concept.

5.1 External Radiation Exposures

There has not been a single instance throughout the history of license SNM-1067 which has resulted in any individual exceeding the 10 CFR 20 quarterly limit of 1.25 Rem. A statistical summary has been provided herein as Figure 5.1 which indicates more individuals have been falling into the lower exposure categories over the past 3 years. The slight increase in the higher categories can be attributed to the higher throughput of uranium in the past 3 years (112 MTU in 1977 to 157 MTU in 1979) and the higher enrichments being processed as utility demand for extended-life cores increases. (License SNM-1067 was amended in 1988 to allow an increase in the maximum allowable enrichment to 5.0wt% U235).

5.2 Internal Radiation Exposures

The most accurate results concerning actual internal deposition of radionuclides are found in bioassay results. The accuracy of these results far exceeds the accuracy obtained from personnel breathing zone air samples (BZ's), since BZ samples serve only as an immediate aid in assessing internal exposure potential and do not conclusively indicate that the material was actually ingested.

During the past 3 years, all urinalysis results were less than 1 microgram U/liter (the lower limit of detection for our fluorimetric method of analysis) with 3 exceptions in 1978 where individuals' results were 2, 3, and 4 micrograms U/liter respectively.

In-vivo lung counting results over the past 10 years clearly indicate that no individual has ever received a maximum permissible lung burden (MPLB), which is about 200 ugm U235 for low enriched uranium. Due to the extreme sensitivity required to detect such low amounts of U235, most results are reported as zero with a statistical accuracy associated with them. Figure 5.2 is a summary of lung counting for the past 5 years and clearly indicates that the ALARA concept practiced at C-E has paid off.

5.3 General Air Sampling Results

Results in graphic form are shown for all air sampling stations in Figure 5.3 thru 5.8. All results are below 15% of MPCa for restricted areas (0.0000000001 uCi/cc) and most results are well below 10% of MPC. The only results which were slightly over 15% of MPC are explained below:

2) March 1980 - Powder Prep Station #1 and Presses 1 and 2 (Figures 5.3 and 5.4). The pellet shop had been undergoing a complete cleanup prior to an enrichment change at this time. Our license action limit was exceeded four (4) times. an investigation revealed that the sample heads were colliding with a nearby vacuuming hose which was highly contaminated.

1) June 1978 - Powder Prep Station #1 (Figure 5.3). Several powder-lot cleanups caused rapid plugging of hood filters and reduced ventilation at the time. Problems were also encountered with the granulator screen and the belt dryer.

K-E

46 1323

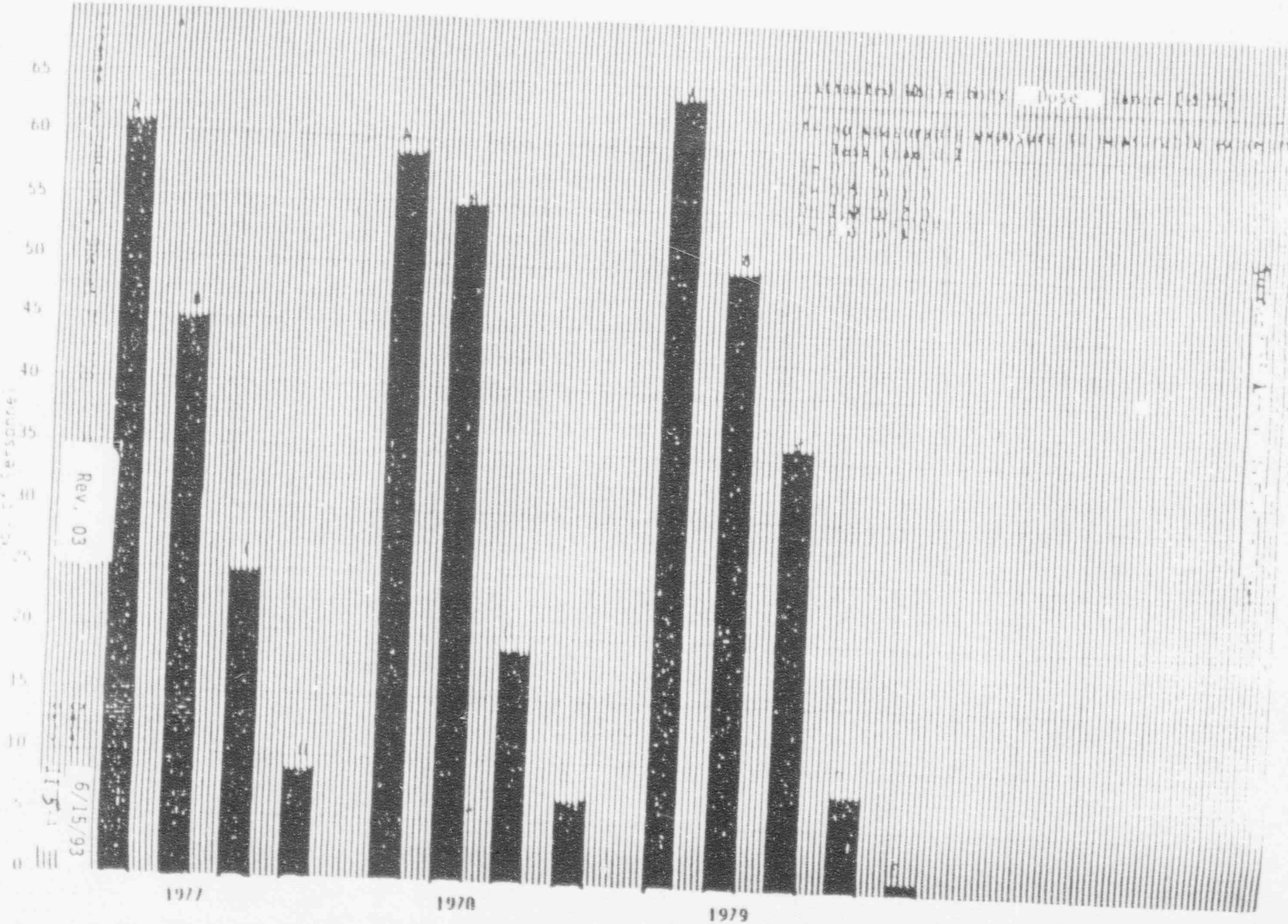
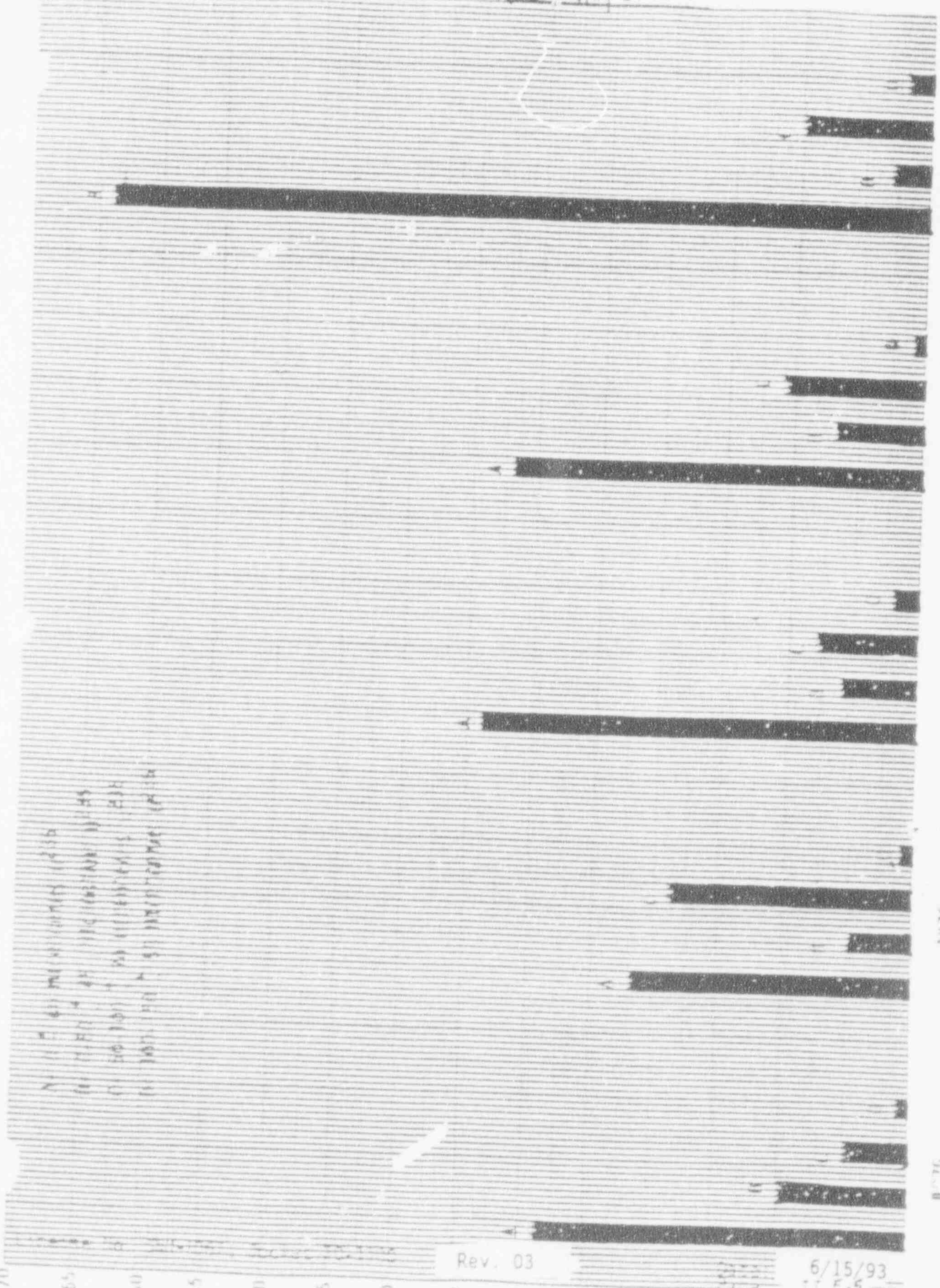


Figure 5:
In-Vivo Lung Counting S...



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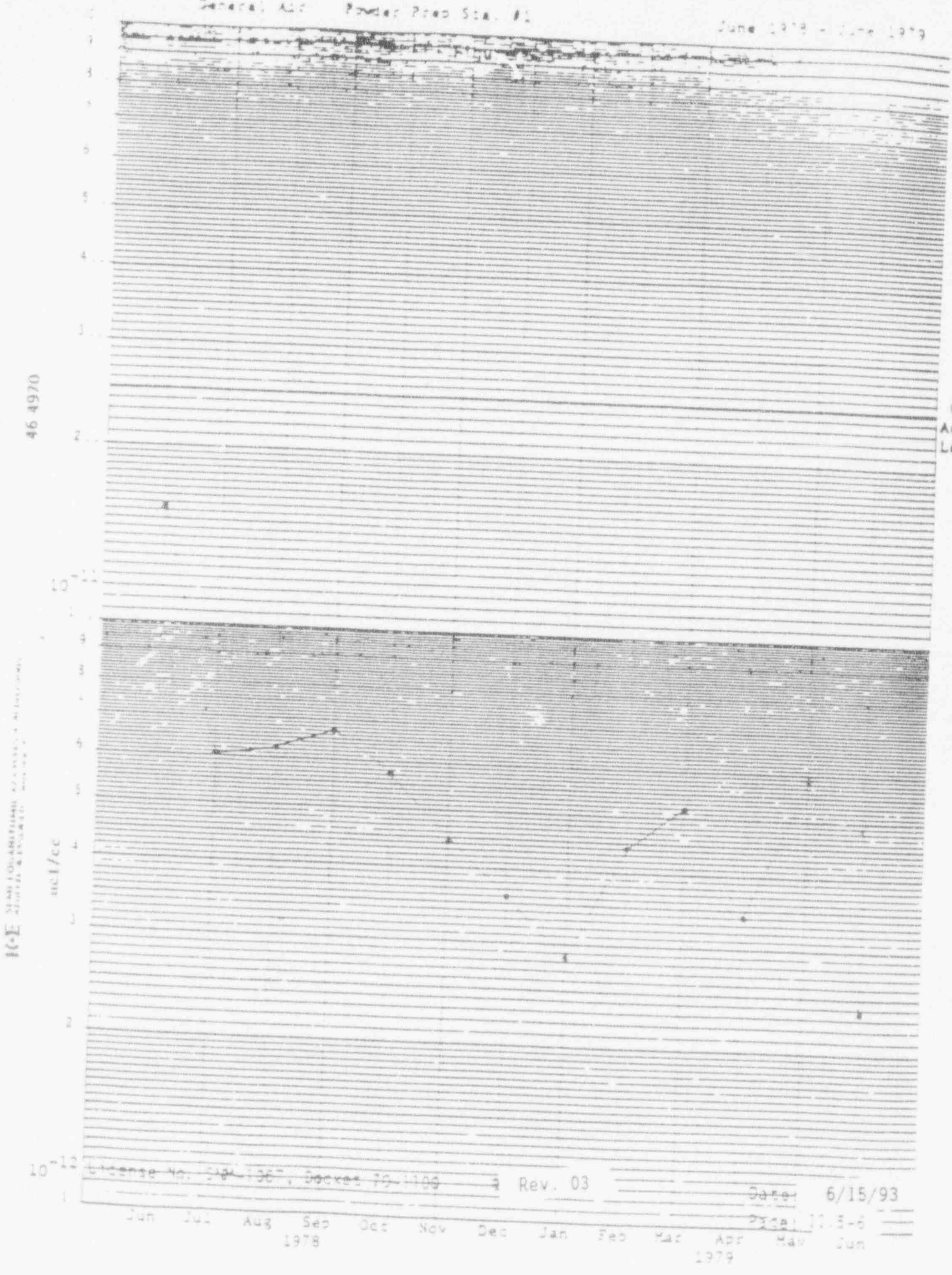
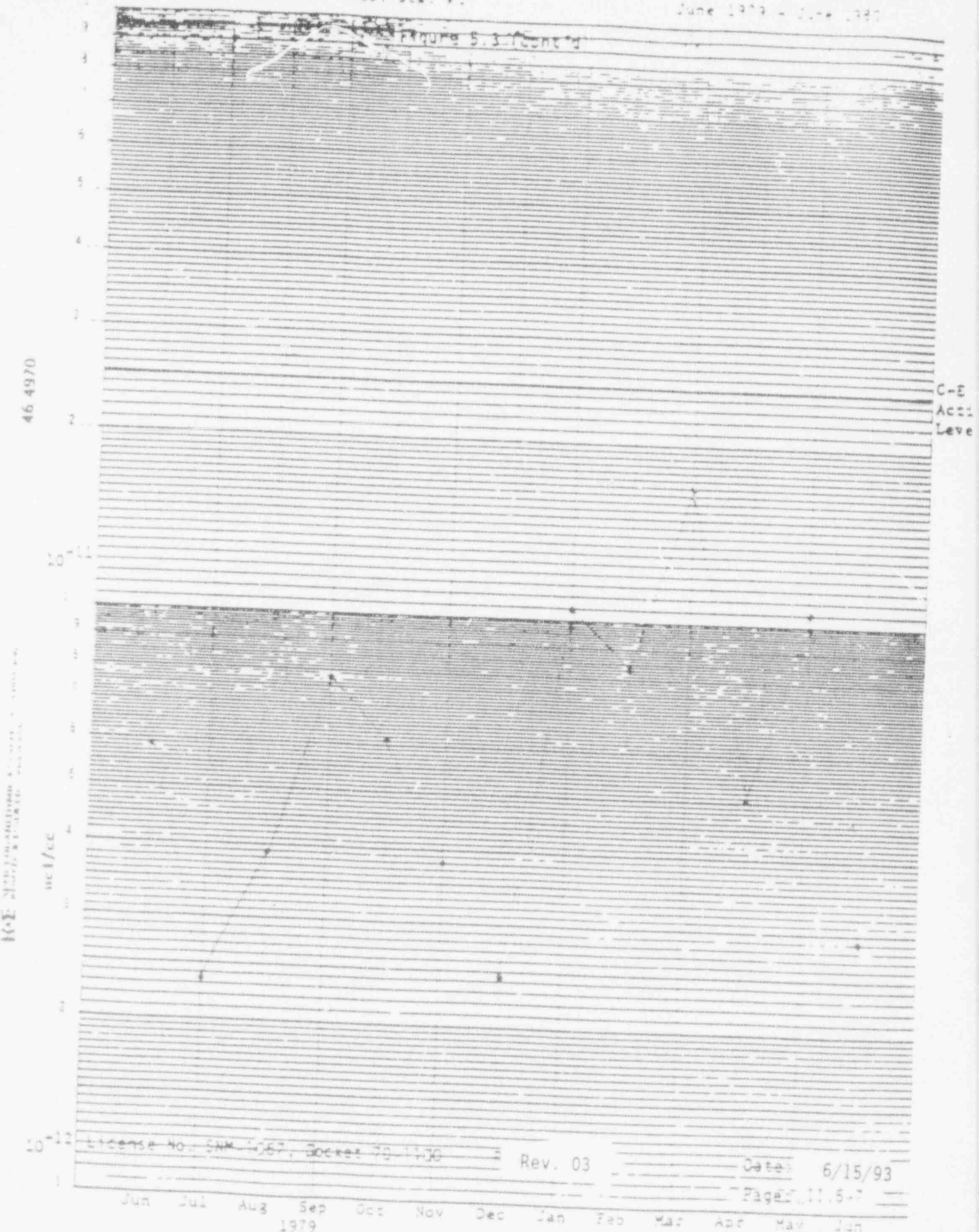

General Air Powder Prep Sta. #1
June 1978 - June 1979
C-E
Acti
Leve
uCi/cc
10^{-11}
10^{-12}
Jun
Jul
Aug
Sep
Oct
Nov
Dec
Jan
Feb
Mar
Apr
May
Jun
1978
1979

General Air Powder Prep. Sta. #.

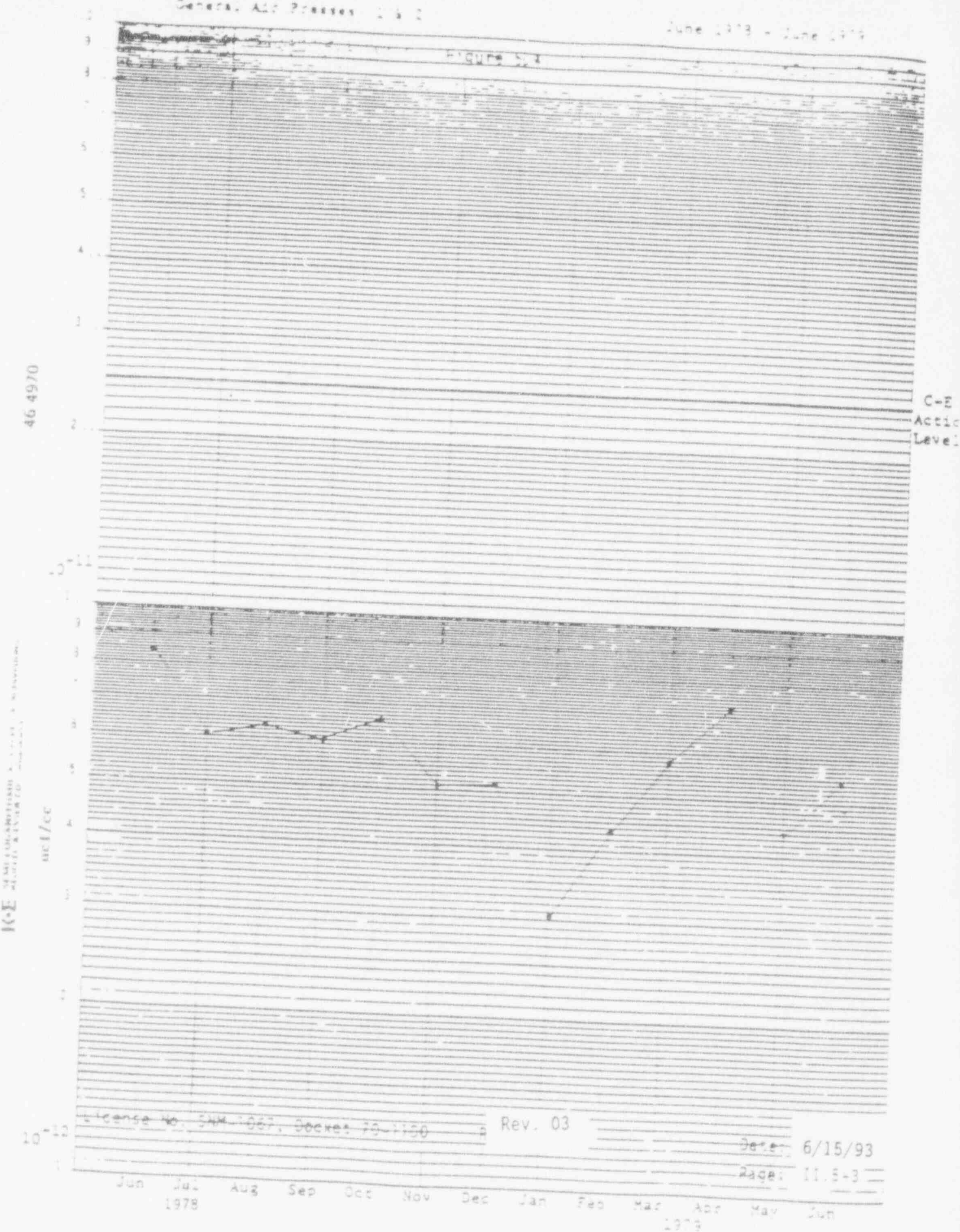
Figure 5.3 (cont'd)



General Air Presses 1 & 2

June 1978 - June 1979

Figure 5.4



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Date: 6/15/93

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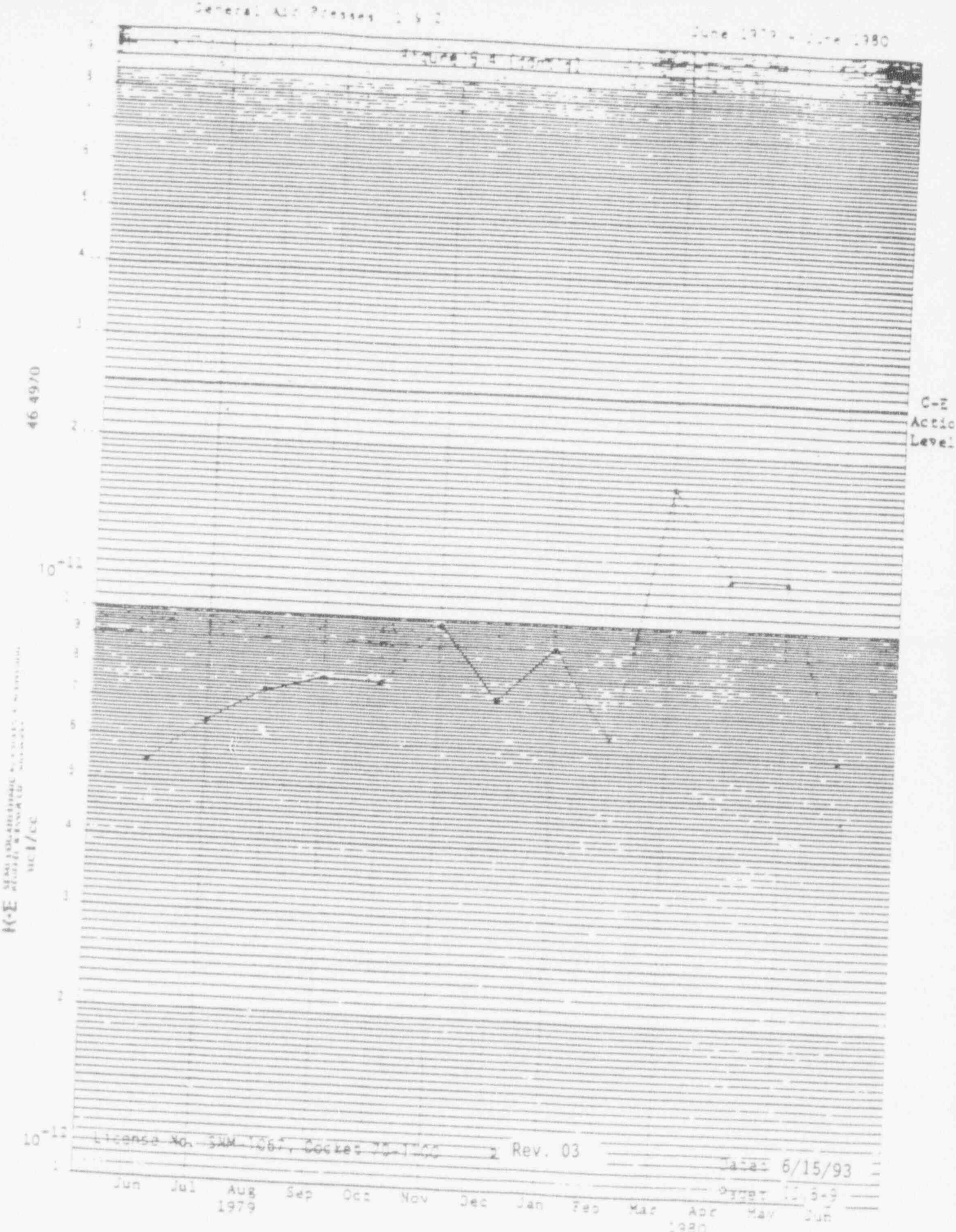
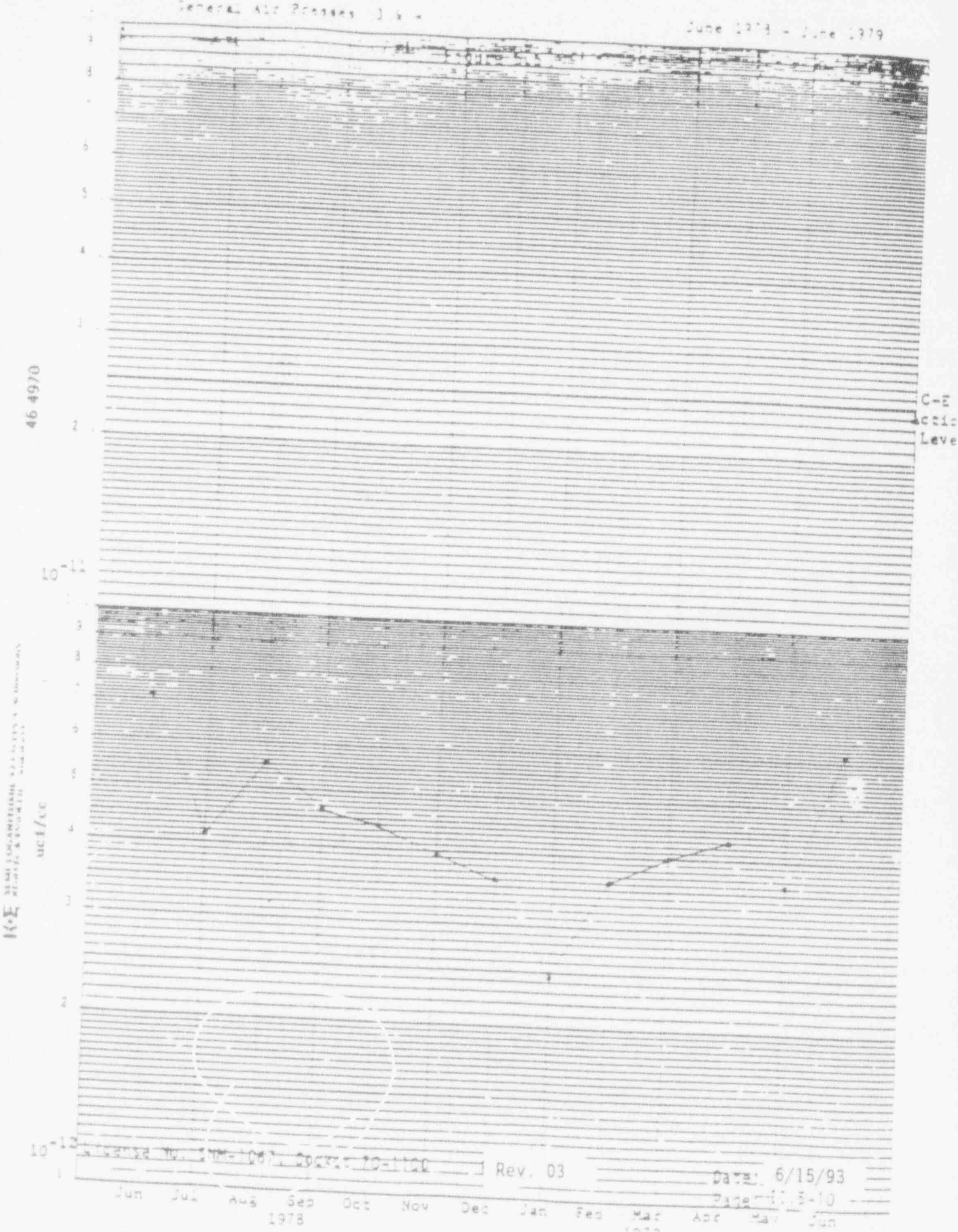
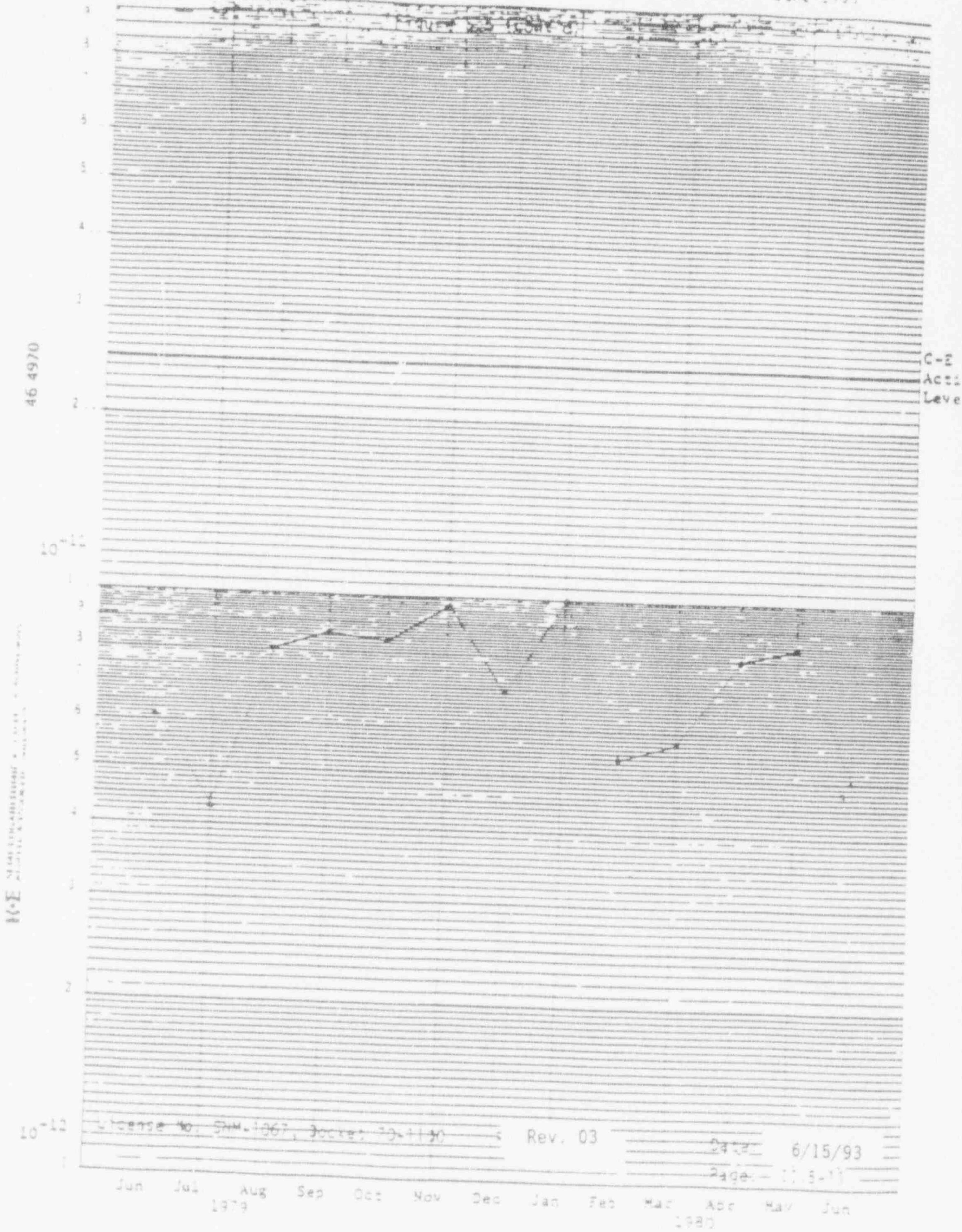


General Air Presses 1 & 2
June 1979 - June 1980
Figure 5.5 (cont'd)
uCi/cc
10^{-11}
10^{-12}
C-Z Actio Level
Jun
Jul
Aug 1979
Sep
Oct
Nov
Dec
Jan
Feb
Mar
Apr 1980
May
Jun


General Air Process
June 1978 - June 1979
uCi/cc
10^{-11}
10^{-12}
C-E
Action
Level
Jun
Jul
Aug
Sep
Oct
Nov
Dec
Jan
Feb
Mar
Apr
May
Jun
1978
1979

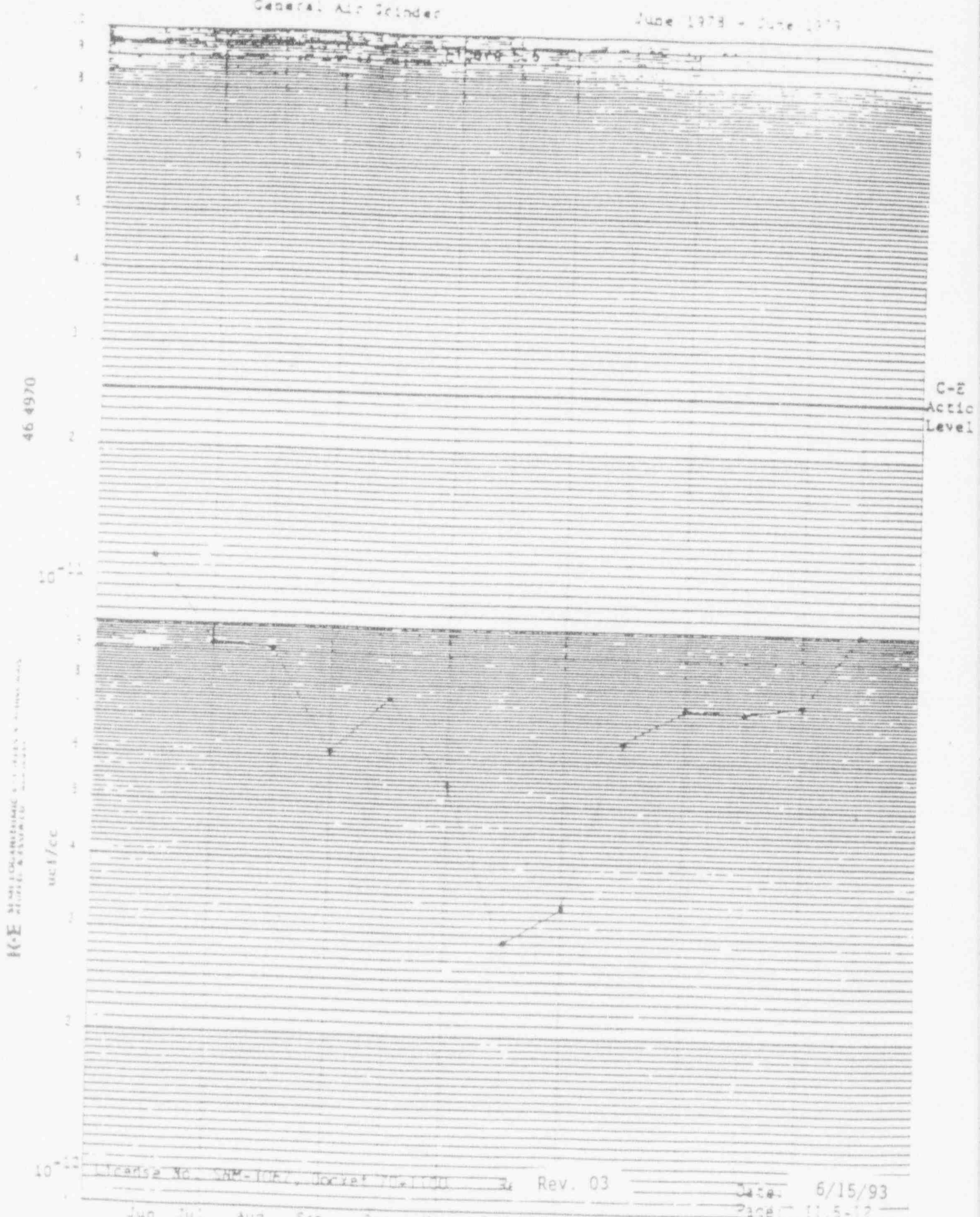
General Air Presses 3 & 4

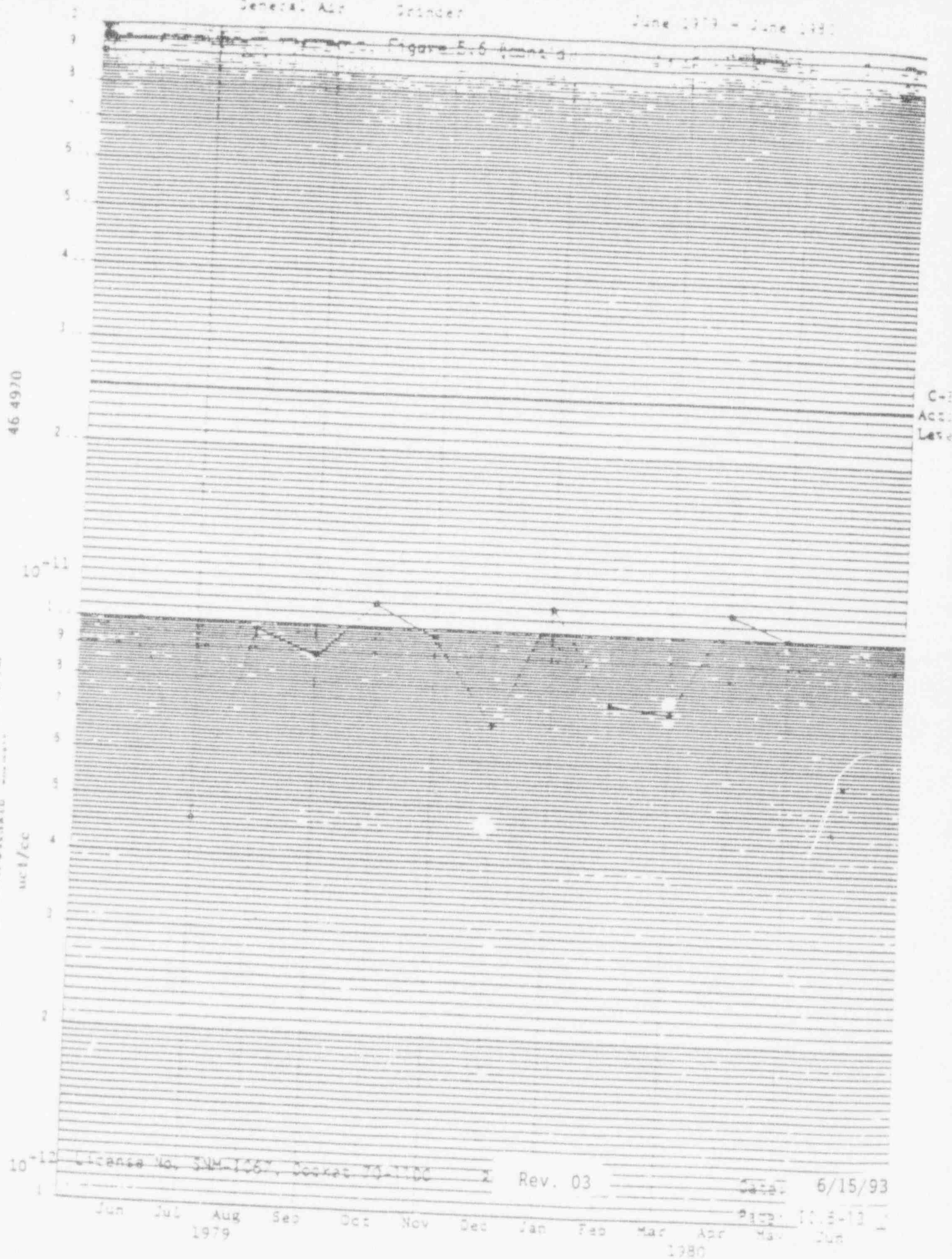
June 1979 - June 1980

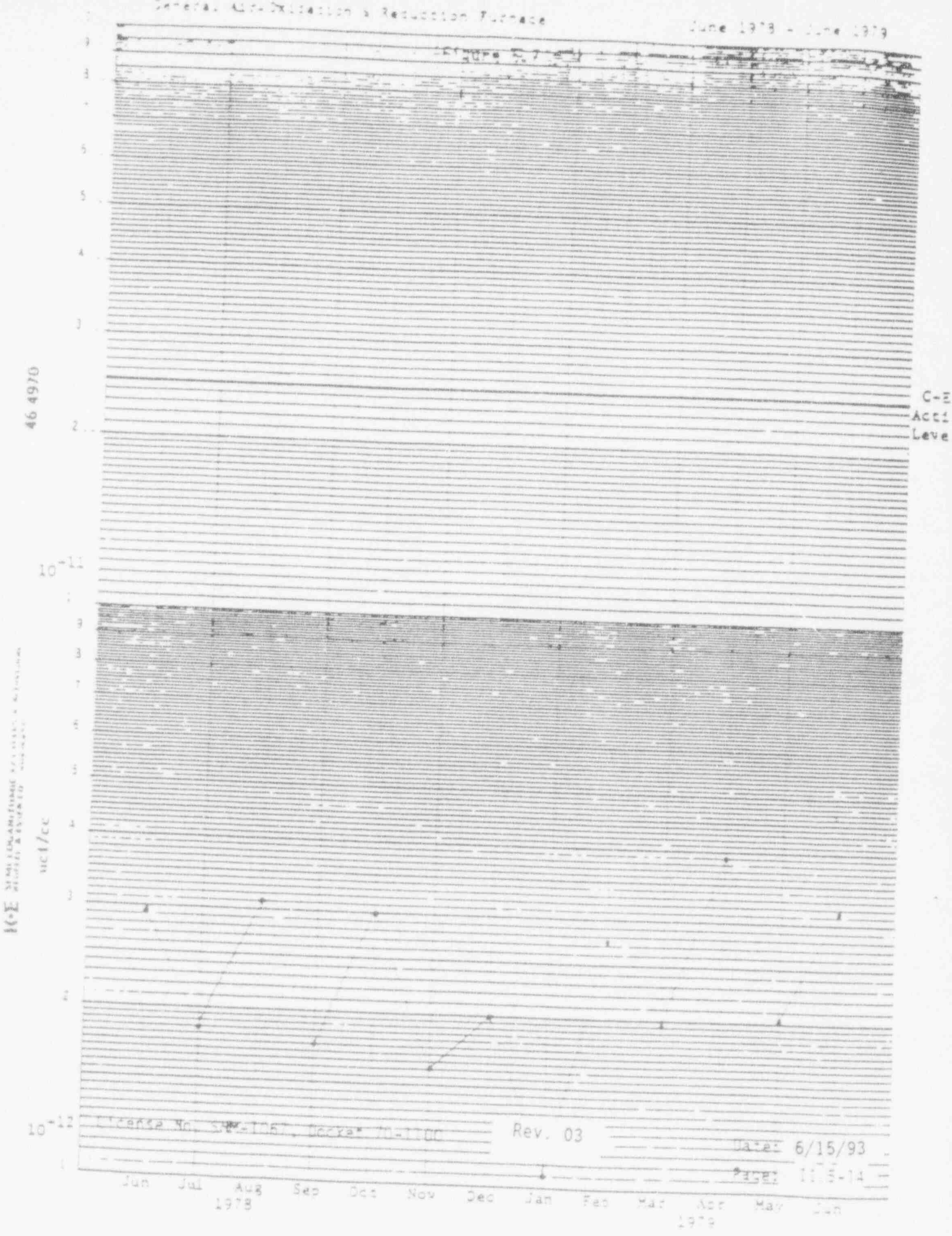


General Air Cylinder

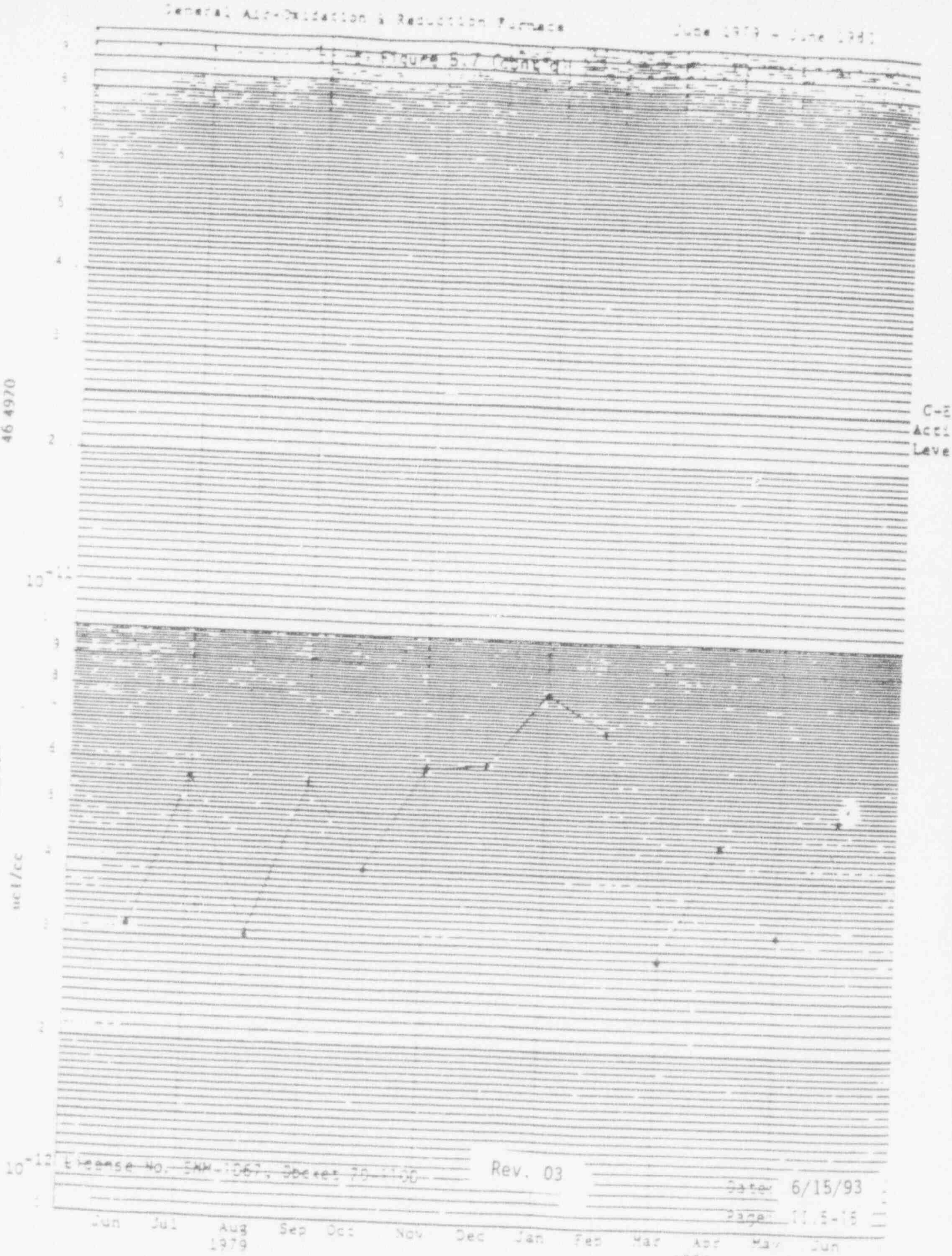
June 1978 - June 1979

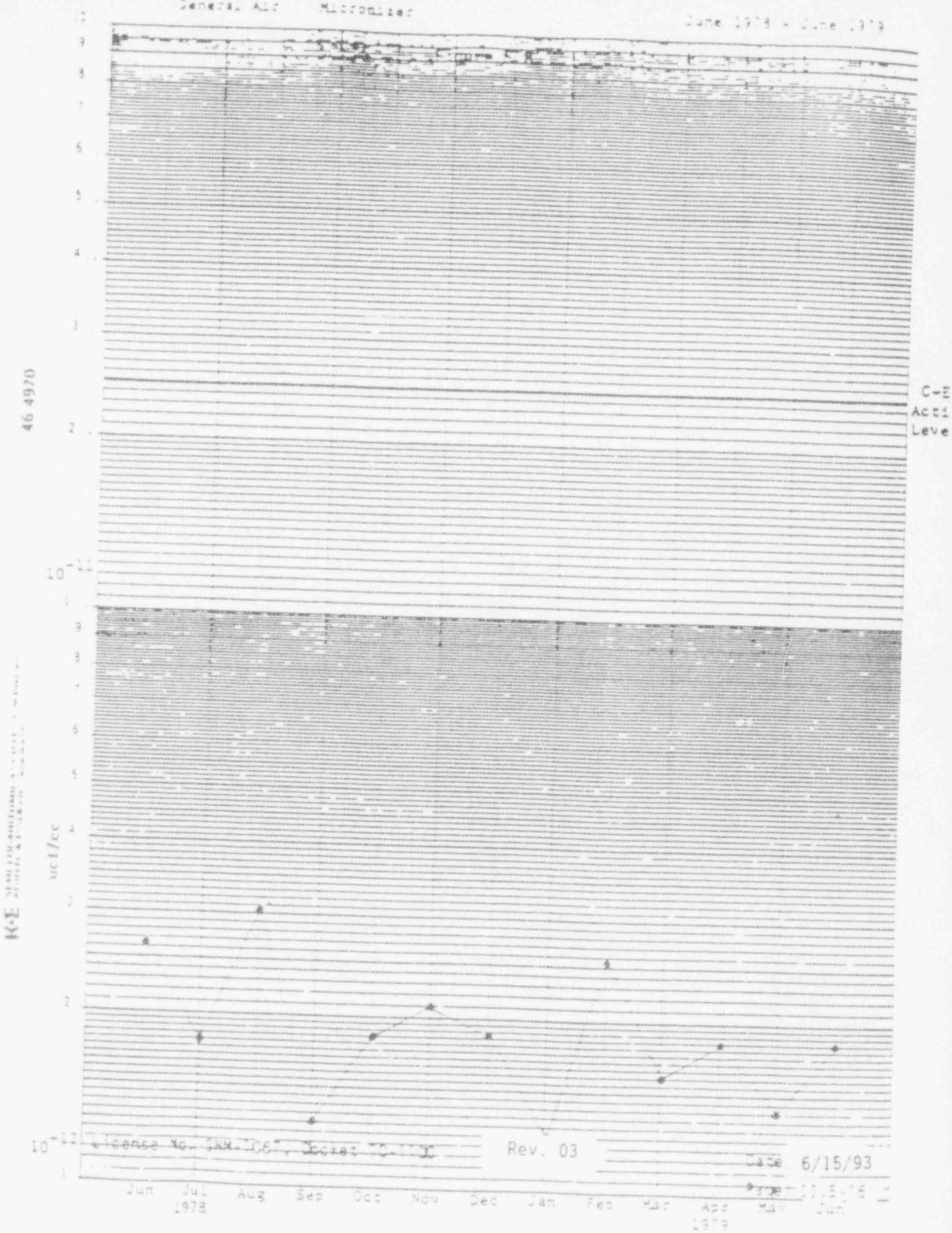



General Air Grinder
June 1979 - June 1980
Figure 5.6
46 4970
C-E
Action
Level
10^{-11}
10^{-12}
uci/cc
Jun
Jul
Aug
Sep
Oct
Nov
Dec
Jan
Feb
Mar
Apr
May
Jun
1979
1980


General Air-Oxidation & Reduction Furnace
June 1978 - June 1979
Figure 5.7-
10^{-11}
10^{-12}
uCi/cc
C-E
Acti
Leve
Jun
Jul
Aug
Sep
Oct
Nov
Dec
Jan
Feb
Mar
Apr
May
Jun
1978
1979
46 4970

II.5, Figure 5.7 (Cont'd)




General Air Micronizer
June 1978 - June 1979
uCi/cc
10^{-11}
10^{-12}
C-E
Acti
Leve
Jun
Jul
1978
Aug
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Oct
Nov
Dec
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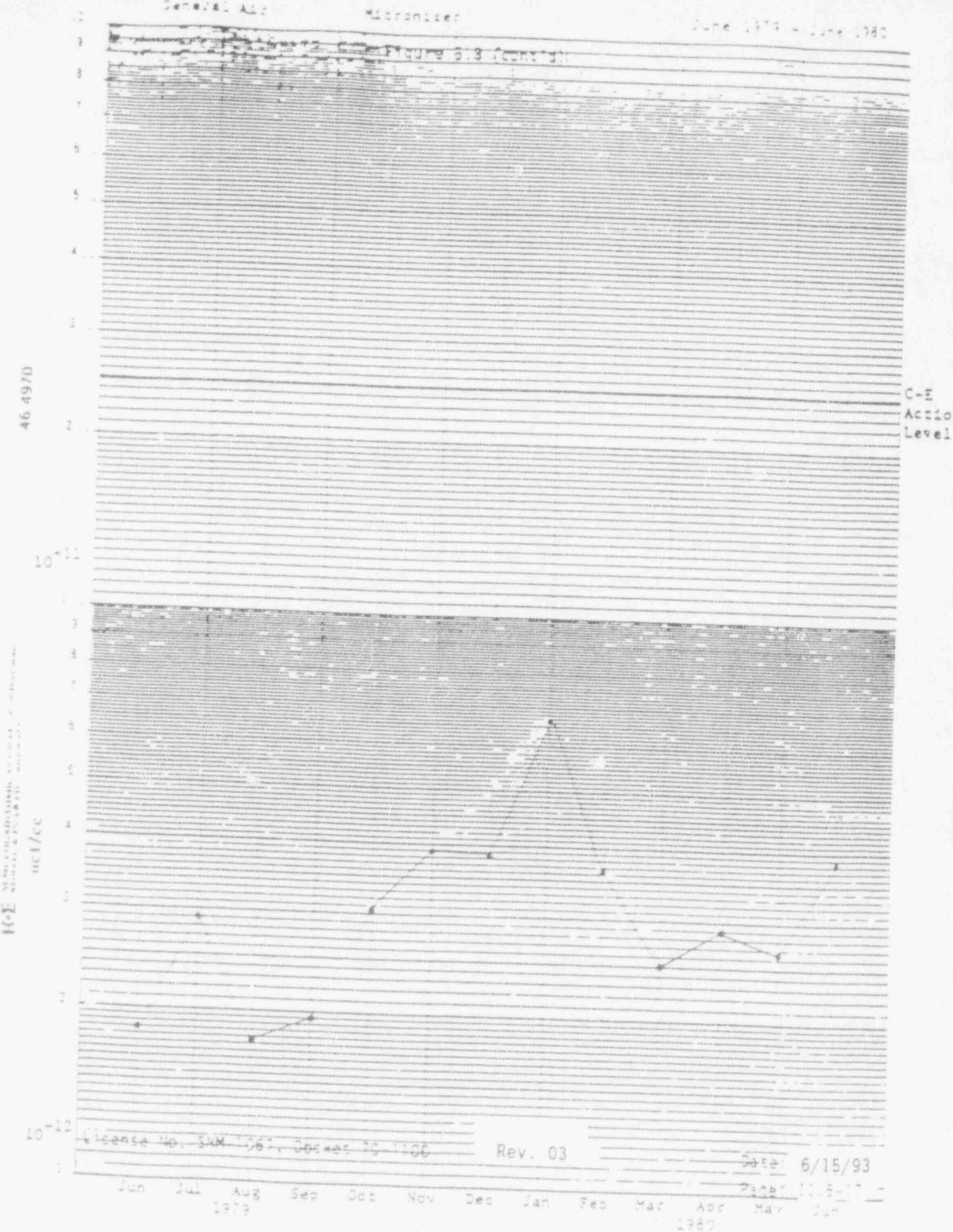
Mar
Apr
1979
May
Jun
46 4970
K+E

Figure 5.3 (cont'd)



6.0 ENVIRONMENTAL SAFETY

6.1 Liquid Effluent Discharges

All liquid wastes, in-process, and clean-up rinse water solutions are sampled to verify that HPCW is not exceeded, and are then introduced t any one of ten 2,000 gallon retention tanks, located in the Liquid Waste Building #6. Sinks and showers in the laboratories and the manufacturing facility are also drained to these retention tanks and provided additional dilution. Before these tanks are discharged to th site creek, which flows into the Farmington River, a 500-ml sample is withdrawn and forwarded to the Radiochemistry Labortory for gross alph and beta analyses. Water is discharged to the environment at, or belo 0.000003 uCi/ml (this is ten percent of MPC W for insoluble uranium). The allowed discharge level for unidentified mixtures of radionuclides is 0.000000003 uCi/ml. (This is ten percent of MPC W for unidentified mixed radionuclides). Where the levels of activity exceed these limits, the water is diluted before being discharged. The instruments measuring the liquid waste level in each dilution tank shall be calibrated on an annual basis.

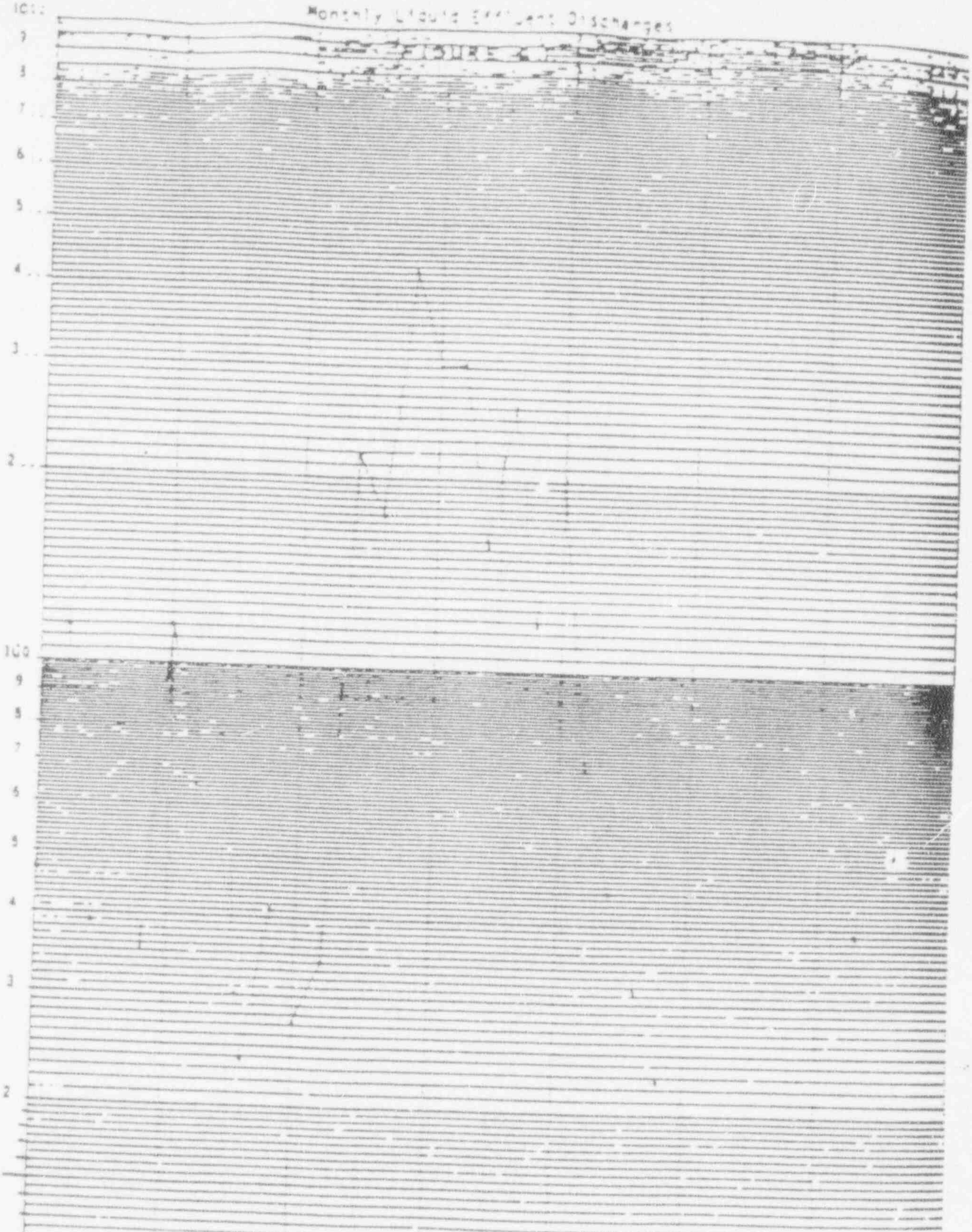
Monthly liquid effluent discharges are shown graphically in Figure 6.1 for the past 3 years. The increase in the amount of uranium being discharged is due in part to the increased throughput in the manufacturing facility.

6.2 Airborne Radioactivity Discharge

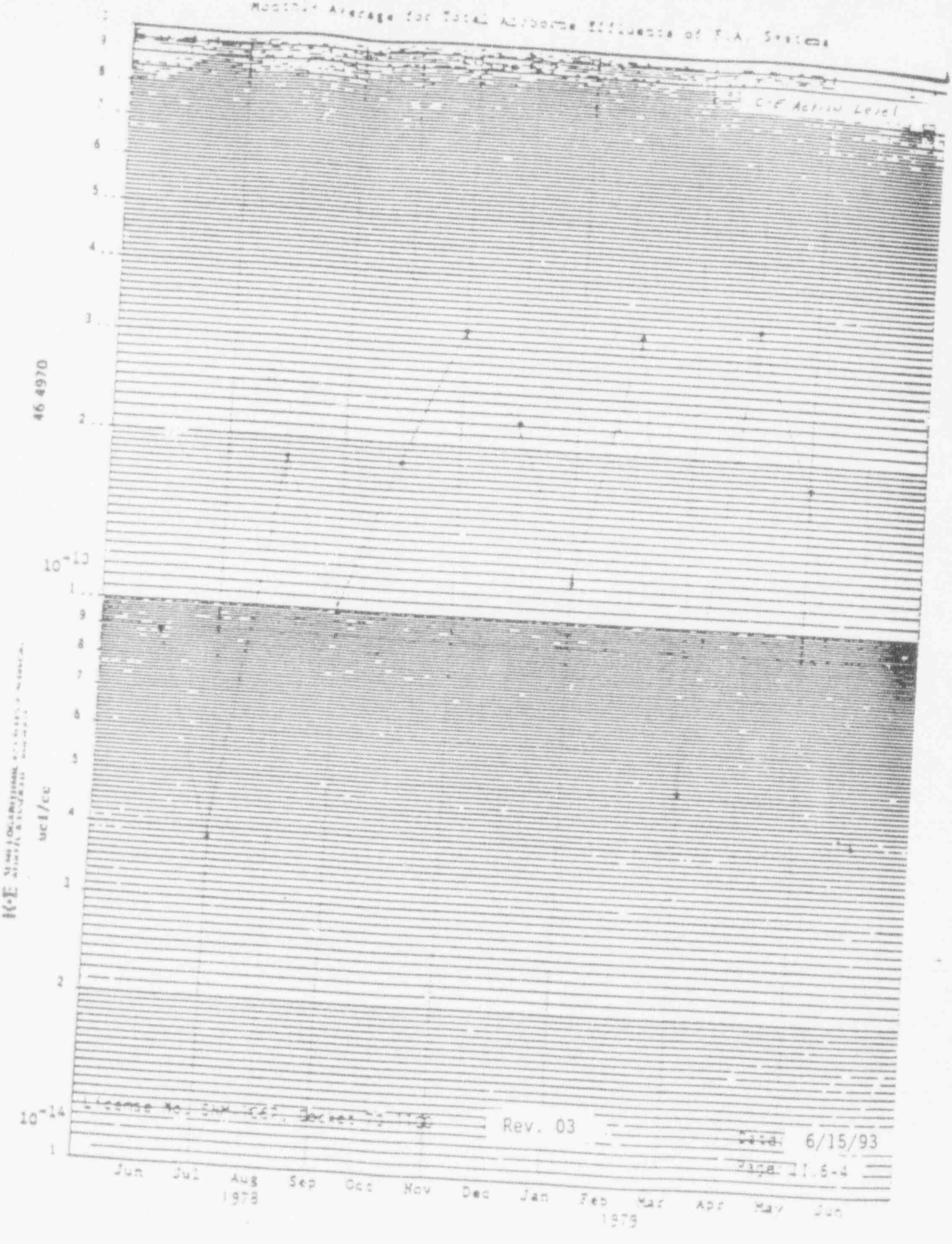
All airborne releases are monitored continuously following double HEPA filtration. Samples are analyzed daily and totaled weekly and monthly for all four ventilation systems. The monthly average concentration in uCi/cc for all 4 systems is shown graphically in Figure 6.2 for the period 6/78 - 6/80. All levels are less than 10% of MPCa for unrestricted areas with the exception of June 1980. During this month, a HEPA filter change was performed on the FA-4 system where the filters were not sealed properly. After discovery of this higher than normal discharge, corrective actions were taken to preclude recurrence.

Monthly Liquid Effluent Discharges

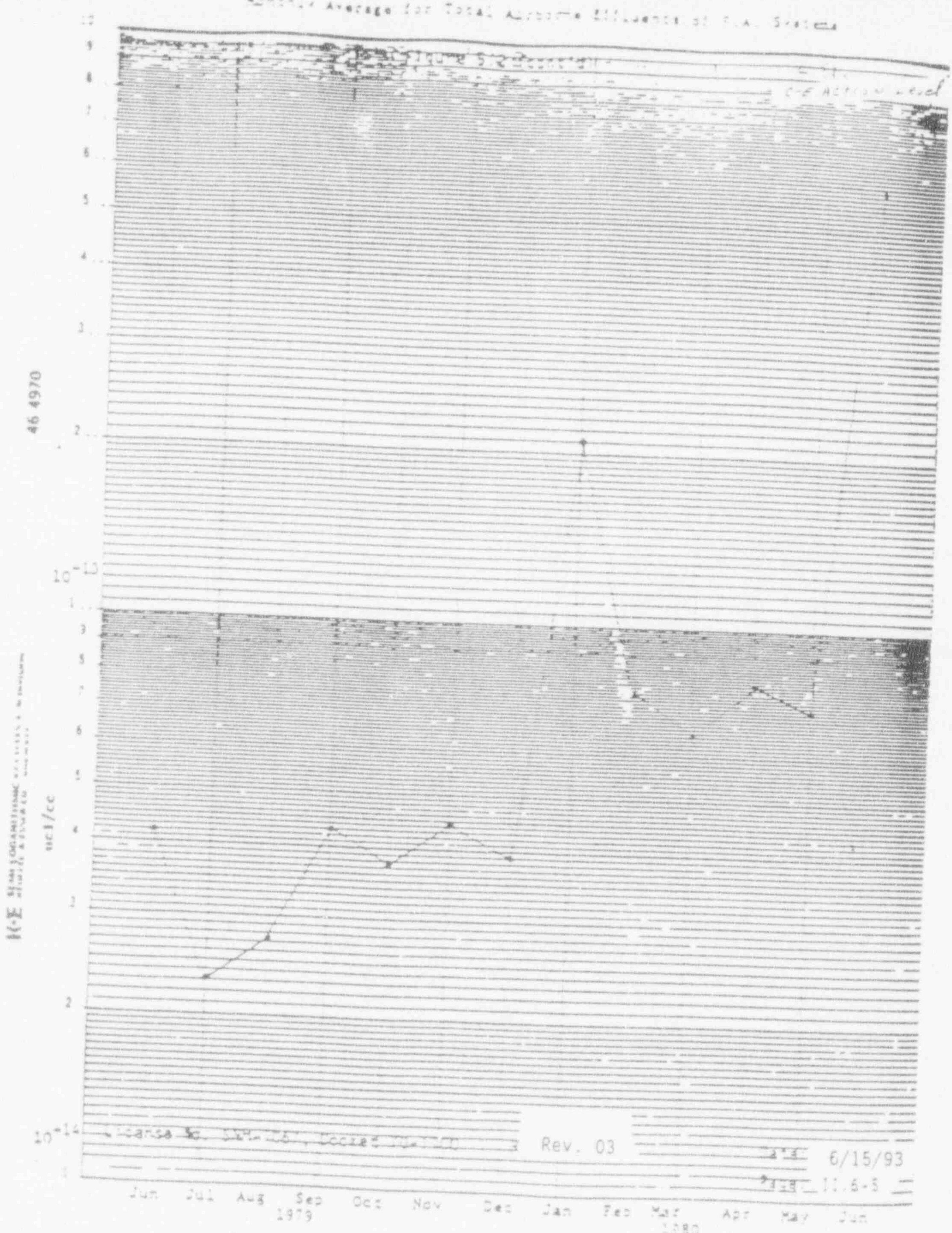
FIGURE



Monthly Average for Total Airborne Effluents of F.A. Systems



Monthly Average for Total Airborne Effluents of F.A. Systems



7.0 NUCLEAR CRITICALITY SAFETY

7.1 Use of Surface Density Technique

7.1.1 Use of Surface Density Criteria of 50% for Mass Limited Units on Single Levels

Mass limited units as specified in Table 4.2.5 are to be spaced to a maximum array surface density of 50% of the optimum critical surface density based on mass per unit area. The criteria is supported as follows: Consider the following analysis at a infinite planar array of 5 gallon steel containers containing 35 kilograms of 5.0 wt % U235 with the remaining volume of the 5 gallon container filled with water. The UO_2 water mixture is assumed to be uniformity distributed within the container. The 5 gallon steel container is 14.25 inches high, with a bottom diameter of 10.25 inches and a top diameter of 11.25 inches. In this analysis, this container has been modeled as a carbon steel cylinder with a diameter of 10.75 inches, with a wall thickness of 0.0275 inches. A 12 each reflector was placed above and below the array. The results of the KENO IV using 16 energy groups are:

Distance Between Containers	K-effective
12 inches	0.9584 ± 0.0065
14 inches	0.9414 ± 0.0081
16 inches	0.9268 ± 0.0070

From this analysis it is concluded that use of a limit equal to 50% of the minimum critical slab surface density (at optimum moderation) express in terms of mass per unit surface area is safe, with a maximum nominal array reactivity of less than 0.95.

Allowable Surface density - The minimum mass surface density for U(5)O2 is calculated to be 10.4 kgU/square foot (Reference 1). The safe limit is therefore 5.2 KgU/square foot.

Spacing Requirement - 20.33 inches for a 35.0 kilogram mass to meet the surface density criteria

Reference 1): R. L. Stevenson and R. H. Odegaarden, "Studies of Surface Density Spacing Criteria Using KENO Calculations," Transactions of the American Nuclear Society, 12, 890 (1969).

7.1.2 Double Level Areas

a) Mass Limited Areas

By doubling the required spacing area for mass limited uni in each level, the overall mass surface density is preserved.

b) Geometry Limited Units

Spacing requirements for geometry limited units on two levels are based on 16% of the infinite slab surface densi at optimum geometry. Units are limited to a fraction critical of 0.4. This limit has been validated by two KENO calculations, for 10.7" diameter x 72" long cylinders (f = 0.36). These cases are described below:

Material	- UO_2, 3.5 w/o U235 optimum moderation
Individual Unit Limit	- 10.7" x 72" lg.
Fraction Critical	- 0.36
Array Reflection	- 16 inch thick concrete floor, 4 inch thick concrete roof, 2: feet above the floor.
Array Pattern	- Infinite square pattern on two levels, with 10 ft. vertical separation between levels. In one case, the upper pattern rests on a 1/2" thick steel plate: the other, on a 3/8" thick steel plate.
Array Surface Density	- With units spaced on a 80.4" pitch, the total array surface density (both levels) is 2.36 liter/square foot, or 16% of the critical infinite slab surface density.
Calculated Array Reactivity	- With 1/2" steel, K eff=0.9262 ± 0072. With 3/8" steel, K eff=0.9273 ± 0.0074. With both the upper and lower arrays in contact with the 3/8" steel deck, K eff=0.9223 ± .0070.

7.1.3 Calculation of Fraction Critical

The concept of fraction critical for an SIU is based on an arbitrary definition which ratios the SIU mass, or equivalent spherical mass to that of an unreflected critical sphere of the same material, assuming optimum moderation. Several prescriptions for calculating this value have been described in the literature. Depending on the method selected, somewhat varying results may be obtained.

In evaluating SIU's for this license, nonspherical SIU's are reduced to equivalent spherical shapes using buckling conversions, in conjunction with somewhat arbitrarily selected extrapolation lengths which vary with fissile density and moderation, physical form, and unit shape. For this license, extrapolation lengths are taken directly from Figure 2 of LAMS-2537. As they are consistently used, any bias introduced is consistent, and based on sphere data taken from the UKAEA Handbook, is also relatively minor.

As a further check on the reasonableness of the use of these values for low enrichment uranium, we have compiled critical data from WCAP-2999, DP-1014, and the UKAEA Handbook for UO2, and from LA-3612 for U-metal mixtures and UO2F2 solutions. These data are presented in Table 7.1.3 and show that for the oxide, there is close agreement for the reflected sphere radius, for the material buckling, for the reflected extrapolation length, and that 4-4.5 cm represents a reasonable value for reflector savings. These considerations lend support to the use of the extrapolation lengths as provided herein.

Another variable which must be defined for the license is the unreflected critical mass or volume. These values vary from o author to another. For this license, all unreflected critical sizes are taken from the UKAEA Handbook.

An example of the fraction critical calculations follows:

Consider the 10.7" diameter cylinder limit. From data take from Figure 1.D.14 of UKAEA AHSB Handbook 1, the minimum critical unreflected volume for homogeneous 3.5% U235 (UO2) is 79 liters.

The volume of a sphere having the same buckling as the 10.7 cylinder of homogeneous UO2 is:

$$B^2_{SIU} = \left[\frac{2.405}{\frac{10.7 \times 2.54}{2} + 2.25} \right]^2 = 0.0230\ cm^{-2}$$

$$V_{SIU} = \left[\frac{3.14}{B} - 2.1 \right]^3 \times 4.19 = 26.9\ liters$$

The fraction critical of the SIU is:

$$\frac{26.9\ L}{79.0\ L} = 0.34$$

7.1.4 General Consideratons for 16 x 16 Fuel

Several aspects of fuel handling as they relate to the 16 x 16 fuel design are not specifically evaluated in view of the general observation that the reduced pellet and rod diameters render this material less reactive than the 14 x 14 fuel which has been extensively evaluated herein.

This finding is based on study of DP-1014 and evaluations made in connection with the calculated values reported in Figure 8.25.

7.2 Validation of Calculational Methods for Nuclear Criticality Safety

7.2.1 Validation for Calculations of Heterogeneous Configurations

To validate the methods used in criticality analysis of fuel manufacturing processes, the 2.35 w/o U235 UO2 critical separation experiments by Battelle (Reference 1) were analyzed in three dimensions. The mean K eff value of these nineteen experiments was 1.00157 with a standard deviation of 0.00419.

The experiments are concerned with the critical separation between water-flooded subcritical clusters of fuel rods in the presence of various fixed neutron poisons. The experiments wer carried out in a 1.8m x 3m x 2.1m deep tank provided with features specifically designed and built for these experiments. These experiments involved aluminum-clad 2.35 wt% U235 enriched UO2 rods about 12mm in diameter by 914mm in length. The critical separation between three subcritical clusters of these rods aligned in a row was determined and analyzed with and without the following neutron absorber materials (neutron poisons) located between the clusters: 304 L Stainless Steel with 0, 1.05 and 1.62 wt% boron; and boral.

Description of Experiments

The experiments analyzed each consisted of three assembly-like configurations separated by water and/or poison plates with the spacing adjusted to criticality. Figure 7.2.1 illustrates typical top and end view of the arrangements. The 914mm length fuel rods 11.176mm in diameter of 2.35 w/o U235 in UO2 clad with 6061 aluminum having an O.D. of 62.7mm and 0.762mm thick with different alloys of aluminum for top and bottom plugs. A fixed square center-to-center pin pitch of 20.32mm was maintained. The number of pins in the width of the cluster varied (in different experiments) between 14 and 17 and the length from 20 to 24 pins. The experimental data on experiments analyzed is given in Tables 7.2.1, 7.2.2 & 7.2.3.

Method of Calculation

The calculational methods which were used are essentially the same as those used to determine reactivity for fuel assembly storage racks, fuel shipping containers, and other fuel configurations found in fuel manufacturing areas: broad group neutron cross sections are based on the CEPAK Code (Reference 2). Using an appropriate buckling value and taking proper 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 29 multi-group type calculations from THERMOS. 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 (Reference 3). 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 (Reference 4) is used. Since however, the short fuel length made necessary a three dimensional treatment, the Monte Carlo Code KENO IV (Reference 5) was used with six axial levels. Batches of one hundred neutron histories were used with the first four discarded. Calculated K eff values are shown in Table 7.2.4. For economy, about 50 batches were run for most cases, however, because of their greater use in fuel storage analyses, about 500 batches were employed for the plain stainless steel and boral experiments.

The mean value of the calculated keff is 1.00157 with a standard deviation of .00419; thus at a 95/95 confidence level using a sigma multiplier of 2.423, the keff values are between 1.012 and 0.991.

References:

1. S. R. Bierman, E. D. Clayton and B. M. Durst, Critical Separation Betweed Subcritical clusters of 2.35 w/o U235 Enriched UO2 Rods in Water with Fixed Neutron Poisons, PNL-2438, October 1977.

2. CEPAK - A Systhesis of the following codes:

   FORM - A Fourier Transform Fast Spectrum Code for the IBM-7090, McGoff, D. J., NAA-SR-Memo 5766 (September 1960)

THERMOS - A Thermalization Transport Theory Code for Reactor Lattice Calculations, Honeck, H., BNL-5816 (July 1961).

CINDER - A One-Point Depletion and Fission Product Program, England, T. R., WAPD-TM-334 (Revised June 1964).

3. J. Adir, S. Clark, R. Froelich, and L. Tody, Users and Programmers Manual for the GGC-3 Multigroup Cross Section Code, GA-7157, July 25, 1967.

4. R. G. Sottesy, R. K. Disney, A. Collier, Users Manual for the DOT-IIW Discrete Ordinates Transport Computer Code, WANL-TME-1982, December 1969.

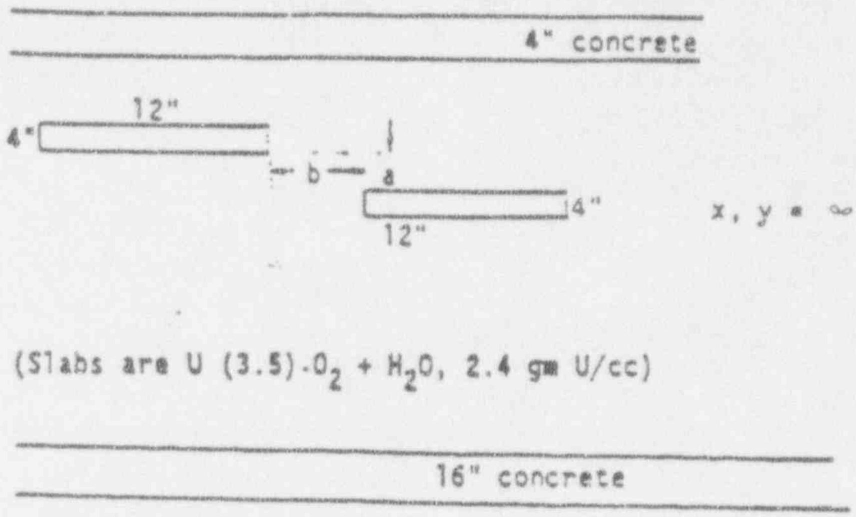
5. L. M. Petrie and N. F. Cross, KENO IV, An Improved Monte Carlo Criticality Program, ORNL-4938, November 1975.

7.2.2 Validation for Calculation of Homogeneous Configurations

Validation of the methods used in criticality analysis of homogeneous fuel configurations is described in a report entitled "Validation of the KENO Code for Nuclear Criticality Safety Calculations of Moderated Low-Enriched Uranium Systems." (Y-1948) by G. R. Handley and C. M. Hopper dated June 13, 1974. If computer codes other than KENO are used , or cross sections other than the 16 group Hansen-Roach set are used, a separate validation study shall be performed to identify the range of applicability for the methods employed.

7.3 Non-co-planar Slabs

An array of slabs has been evaluated using ANASIN code. The configuration is shown below:



Evaluations of this configuration yield the following:

a	b	Δke
0	0	0.0
6	0	+0.022
12	0	+0.029
28	0	+0.039
12	0.75	+0.0139
12	1.5	-0.0009

Accordingly, a 1.5-inch horizontal spacing counters the reactivity increase from a non-co-planar configuration. The change in reactivity for $U(3.5)O_2$ is so small that the effect for $U(4.1)O_2$ would not be significantly different. Additional safety is provided by the requirement that non-co-planar slabs be limited to 12-inch vertical differences and separated by at least 6-inch horizontal spacing.

TABLE 7.1

KENO Calculated Arrays of Low Enriched Uranium Subcrits

Array	Composition	Subcrit	fx	gmU/cc	KgU/ft^2	k_{eff}
1	$U(5)O_2$	cylinder 11.5 cm r x 138.6 cm lg	0.3	2.0	12.7	1.019 ± 0.006
2	$U(5)O_2F_2$	cylinder 11.4 cm r x 152.4 cm lg	0.3	1.0	10.1	1.005 ± 0.009
3	$U(5)O_2$	cylinder 16.0 cm r x 211 cm lg	0.3	5.0	40.1	1.000 ± 0.006

Fraction of critical values taken from Ref. #1

TABLE 7.1.3

Critical Parameters for Optimum Moderated Low Enrichment Uranium

	R_{cr} (cm)	R_{cb} (cm)	B^2_m (cm^{-2})	Lambda (cm)	$R_{cb}-R_{cr}$ (cm)
WCAP-2999(a) 3% UO2	24.60		0.0104	6.3	
DP-1014(b) 3% UO2	24.03		0.0103	6.88	
UKAEA (c) Handbook 3% UO2	24.29	28.79			4.5
LA-3612 (d) 5% U-Metal	17.91	22.16			4.25
LA-3612 (e) 5% UO2F2	22.42	25.69			3.27*

(a) Figure III-15 & Figure III-10

(b) Pages 37 and 57

(c) Figures 1.D.10 and 1.D.14 respectively for reflected and bare volumes.

(d & e) Page 26

* UO2F2, being a solution; has somewhat smaller reflector savings than do oxide or metal systems.

Table 7.2.1

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 wt% ^{235}U ENRICHED UO_2-RODS IN WATER

FUEL CLUSTERS

LENGTH x WIDTH 20.32 mm SQ. PITCH (FUEL RODS)	CRITICAL SEPARATION BETWEEN FUEL CLUSTERS (1) (Xc, mm)	EXPERIMENT NUMBER
20 x 17	119.2 ± 0.4	015
20 x 16	83.9 ± 0.5	005
20 x 16	84.4 ± 0.5	049 (2)
22 x 16 (3)	100.5 ± 0.5	018
20 x 14	44.6 ± 1.0	021

(1) PERPENDICULAR DISTANCE BETWEEN THE CELL BOUNDARIES OF THE FUEL CLUSTERS. ERROR LIMITS ARE ONE STANDARD DEVIATION

(2) RERUN OF EXPERIMENT 005

(3) CENTER FUEL CLUSTER AT 20 x 16 RODS. TWO OUTER FUEL CLUSTERS AT 22 x 16 RODS EACH

TABLE 7.2.2

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 wt% ^{235}U 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	1.62	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

* To distinguish from Experiment #005 of Table 1

TABLE 7.2.3

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 wt% ^{235}U 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 CLUSTERS (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 ONE 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

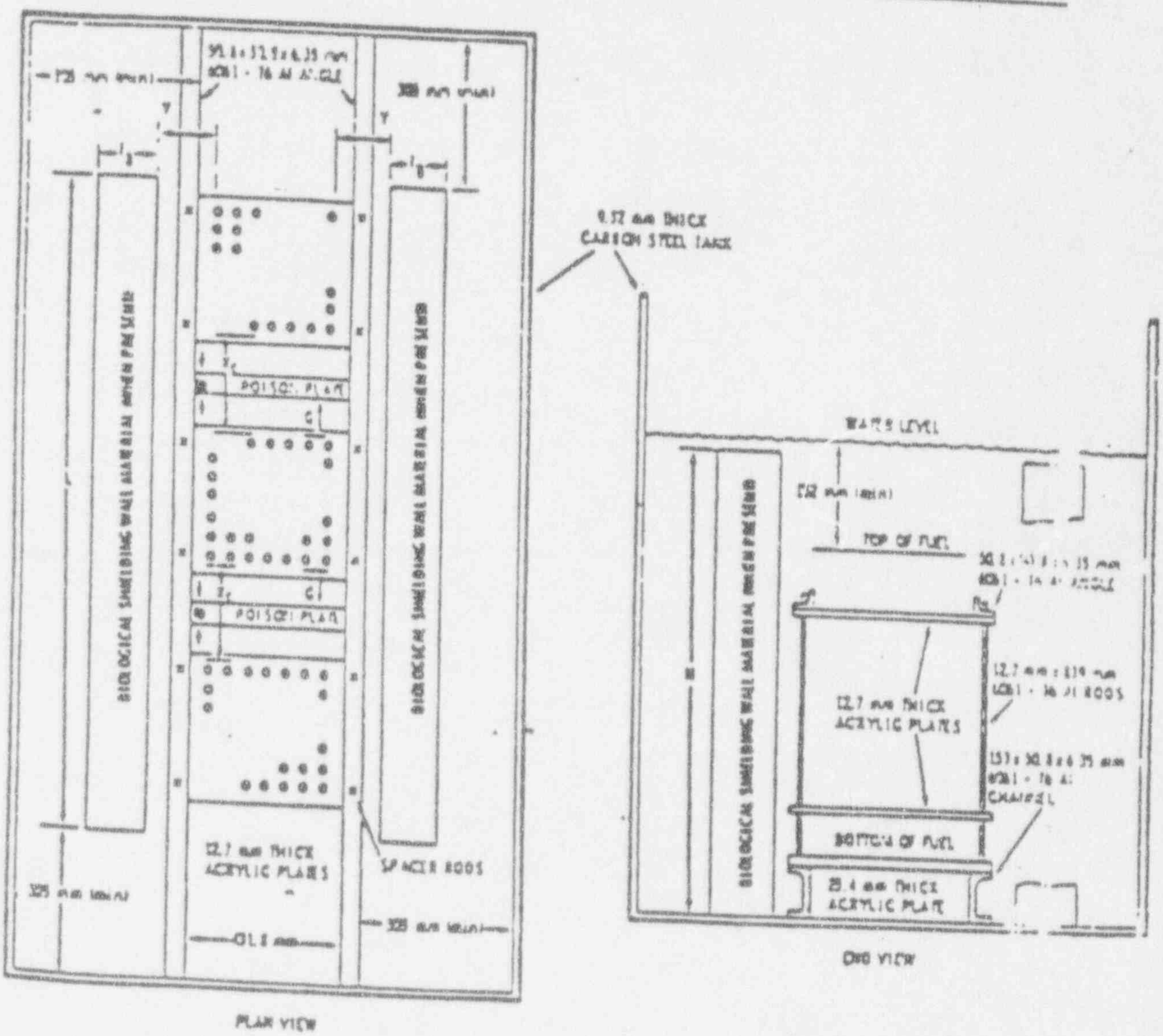
TABLE 7.2.4

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	1.00813	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 0.0 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	0.99793	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	

FIGURE 7.2.1

GRAPHICAL ARRANGEMENT OF SIMULATED SHIPPING PACKAGE CRITICAL EXPERIMENTS



8.0 PROCESS DESCRIPTION AND SAFETY ANALYSES

This section contains detailed descriptions of all operations in the Manufacturing Facility (Building #17 and #21). Sufficient detail is provided to permit an independent verification of the adequacy of the controls for the purpose of assuring safe operations. Nuclear criticality limits are taken from Table 4.2.5 of Part I. In certain operations, the intricacies of the equipment require further analysis, which is provided herein. Details of specific calculations used to support various aspects of this analysis, and several statements and considerations in Section 4 of Part I are discussed in this section. This section provides typical analyses for operations conducted within the scope of this license. Present arrangements of the equipment in the pelletizing facility are shown in Figure 8-1. (Drawing No. NFM-J-4077). This arrangement may be changed in accordance with the procedures in Section 4 of Part I of this license.

8.1 UO2 Processing

8.1.1 Receipt of Material

The as-received 9.75" diameter stainless steel UO2 powder cans, to be stored in the virgin powder storage area (Figure 8.1) and shall be sampled before being placed in the storage area to demonstrate on a 95/95 confidence level that the moisture content of powder lots is less than 5.0 wt%. In addition, all damaged packages where containment is breached will be sampled.

The pellets are received in UNC-2901 shipping containers which may be brought into Bldg. 17 as discussed in Section 4.3.18. Three pallets (six UNC-2901 containers) can be brought into the Bldg. 17 Fuel Pellet Shop Annex and four pallets (eight UNC-2901 containers) can be brought into the Bldg. 17 Pellet Loading Area for storage and unloading. The sealed containers on the pallets can be stored next to each other but must be at least 1 foot from process equipment in the area. Prior to opening a shipping container the pallet must have an area of 21 ft^2 (4.6 ft. x 4.6 ft.) in which the pallet is located. This area will meet the surface density limit specified in Section 4.2 for a mass of 110 kilgrams of UO_2. The pellets in the shipping container are received in 2 inch deep pellet trays with covers. These pellet trays are stored in the pellet storage shelves. The pellets are then treated the same as pellets made in the Bldg. 17 pellet shop.

8.1.2 Virgin Powder Storage Area

The virgin powder storage area is isolated from the remainder of the plant on all sides by concrete block walls, a double steel roof, and a metal fire door. If the door is in the open position, it is automatically closed upon activation of the fire alarm, or on failure of electrical power. The automatic closing feature of this door shall be verified quarterly and records of its performance shall be maintained. These engineered safety features are considered adequate to prevent the introduction of water in the event of a fire. This area will be kept free of combustibles, and located such that there are no potentially hazardous items such as boilers in the vicinity of the area.

Two ammonia crackers are housed in a concrete block building which is located some 25 feet northwest of Building #17. In view of its many redundant safety features, it is not viewed as a potentially hazardous item.

Criticality Safety Analyses

The following assumptions were incorporated into the calculational model of the Virgin Powder Storage Area:

1) All steel structural materials were neglected.
2) The fuel was assumed to be a homogeneous mixture of UO2 containing 5.0 wt% H2O.
3) All storage positions were filled and each individual can was assumed to contain 35.0 kg's of UO_2 at 5.0 wt% U235.
4) No interspersed water moderation was considered.

The KENO-IV Code with sixteen group Hansen-Roach cross sections was used to determine the reactivity of the Virgin Powder Storage Area under the condition noted above. Dimensional details of the model are provided in Figure 8.1. A Keff of 0.7781 ± 0.0043 was obtained for an infinite system in the horizontal direction.

8.1.3 Batch Make-Up

Powder containers are removed from the virgin powder storage area and placed on a conveyor for transfer to the Batch Make-Up Hood. Two 9.75 inch diameter x 11 inch long stainless steel powder containers shall be placed on fixtures in the left side of hood and either a powder container or a 5 gallon pail on the right side of the hood when an appropriate batch of less than 35 Kg UO2 is weighed out and put into a 5-gallon pail. The batch weights and enrichment are recorded on the container. A water tight cover is secured to these batch containers and they are then conveyed to the cone change hood. The cover is placed in the change hood with a water tight blender feed cone and then transferred to the blender hood. The batch make-up operation and the cone change are enclosed in ventilated hoods. Sufficient negative pressure is provided to assure a minimum face velocity of 100 fpm.

Criticality Safety Analysis

The following conservative assumptions were incorporated in the calculational model of the Batch Make-up Hood and Conveyor Change Lift areas:

1. All steel structural materials were neglected.
2. An external mist of .001 g/cc was assumed.

3. The stainless steel powder make-up cans (9.75" diameter and 11" long) on the conveyor were modelled as a single cylinder in the horizontal direction based on the can containing a homogeneous mixture of 35 kg UO_2 at 5.0 wt % U235 and 5 wt % H_2O. This mixture was assumed to be uniformly distributed within the can.
4. Two stainless steel powder cans and two 5-gallon stainless steel buckets (10.75" diameter and 14.25" high) in the batch make-up hood were assumed to each contain a homogeneous mixture of 35 kg UO_2 at 5.0 wt % U235 with maximum moderation. The batch make-up hood was assumed to be covered with a 0.25" film of water.
5. The 5-gallon stainless steel buckets on the conveyors and in the cone change hood and hopper lifts are assumed to contain homogeneous mixture of 35 kg UO_2 at 5.0 wt % U235 and 5 wt % H_2O. This mixture was assumed to be uniformly distributed within the can. It has been assumed that there is no distance between buckets on the conveyors. A single 5-gallon bucket of UO_2 has been assumed in each hopper lift area and three 5-gallon buckets of UO_2 have been assumed in the cone change hood. All 5-gallon buckets in the cone change hood, the vertical conveyor, and in the hopper lift areas are assumed to be covered with a 0.25" film of water.

The KENO-IV code with sixteen group Hansen-Roach cross sections was used to determine reactivity of the Batch Make-up and Conveyor Change Lift areas under the conditions noted above. A Keff of 0.7940 $\pm$ .0053 was obtained for an infinite system in the x and y directions.

Dimensional details of the calculational model are shown in Figure 8.2.

8.1.4 Powder Preparation and Blending

UO_2 powder from one sealed batch container (moderation control assured) is transferred to a blender where it is mixed with a binder. Two separate blenders feed a common powder spread funnel by a means of individual powder transfer pipes entering at a 45 degree angle. An identical powder preparation line runs parallel to this one at a centerline distance of 13 feet. The blending operation is enclosed in a ventilated hood. Each 15.2 liter blender is charged with 34 Kg of UO_2. The total water content of 7 wt% is blended into the powder to form agglomerates. The UO_2 is pre-dried in the blender for 30 minutes, minimum, with hot air prior to transferring onto the dryer belt. The powder after being dried has a water content of less than 5 wt%. At 9 wt% water the blender UO_2 is a slurry, not an agglomeration mixture. The slurry is not transferred onto the dryer belt. It is either processed as recycle material or is dried in the blender.

Sufficient negative pressure is provided to assure a minimum face velocity of 100 fpm.

8.1.4.1 Drying

Agglomerated UO2 powder is spread onto the dryer belt from the powder spread funnel to a controlled depth of 1/2". A complete enclosure is provided around the dryer belt assembly and this enclosure is maintained at a slight negative pressure. The discharge end of the dryer belt utilizes a wiper blade to prevent the flow of significant amounts of material to the plenum under the belt. Nevertheless, the wiper blade and plenum shall be inspected once per week to assure that the wiper blade is functioning properly and that fuel is not accumulating in the plenum below the belt. Records of these inspections are maintained. The belt dryer operates on a 1/2" slab limit. The criticality safety analysis also assumed an accidental accumulation of up to 1/2" of powder under the dryer belt. The wiper blade scrapes the powder from the belt into the granulator. The wiper blade is inspected weekly as well as the UO_2 depth under the belt. If the accumulation exceed 1/2" it is cleaned up. Even if the above was not done, the accumulation under the belt would not exceed 3.5 inches due to the physical space limitations. The 3.5" below and the 1/2" above which meets the slab thickness limits in Table 4.2.5. The safety of the dryer assembly is assured by this restricted slab thickness. The granulator controls are wired to the motor control such that the dryer belt cannot be activated unless the granulator is turned on. An over temperature condition will shut off the heating elements and light a warning light.

8.1.4.2 Granulation

Dried oxide is gravity-fed into a granulator where it is sized for subsequent pressing. The granulated powder is then gravity-fed through a discharge funnel ending in a 2 inch square opening. A short adapter of 2 inch circular cross section is welded to the funnel to allow connection of a 2" diameter hose which is then connected to a portable hopper below.

A complete enclosure is provided around the granulator screening mechanism. The enclosure contains a level probe which will shut off the drier belt and heaters should the granulator discharge funnel fill with UO_2 powder. It is maintained at a negative pressure to preclude dusting.

Criticality Safety Analysis

The powder blending, drying, and granulation stations were divided into two parts for calculational purposes. The back end of the stations included the blenders, the powder transfer pipes leading to the powder spread funnel, and the first 10 feet of the 30" wide dryer belt. The spread funnel is fixed in position to restrict the powder discharge from it to a 24" wide and 1/2" deep layer of UO_2. The front end of the station included the last 10 feet of the 30" wide dryer belt, the granulator, the discharge funnel and hose and the large cylindrical press feed hopper.

The following conservative assumptions were incorporated into both calculational models:

1. An external mist of 0.001 g/cc was assumed.

2. The UO2 powder was assumed to have a density of 3.5 g/cc at an enrichment of 5.0 wt% U235 and a water content of 15 wt %, which is the water content if two times the normal amount is placed in the blender.
3. Any structure containing UO2 powder was assumed filled to capacity.
4. Although the belt dryer is limited by a mechanical feeder to 1/2" of UO2 powder, the model allows for an accidental accumulation of 1/2" under the dryer belt in the event of malfunction of the wiper blade.
5. All surfaces of the structures containing UO2 powder and the surface of the powder in the blender hood and on the dryer belt were assumed to be covered with a 0.25" film of water.
6. The concrete floor and ceiling have been accounted for in the models.
7. An infinite array of stations was analyzed although there are only two parallel stations.

In the analysis of the back end of the stations it was assumed that the blender hoods are restricted to a maximum of 35 kg UO2 per station, that this mass of UO2 was located at the base of each blender hood directly above each powder transfer pipe and was hemispherical in shape.

In the analysis of the front end of the stations it was assumed that the large hopper (11"OD x 40"L cylinder) contained the UO2 powder at 5 wt % U235.

Sixteen-group Hansen-Roach cross sections were used in KENO-IV to determine the reactivity of the system. Based on the conditions described above, the following Keff were obtained.

back end of station Keff = 0.5164 ± .0057

front end of station Keff = 0.8372 ± .0061

Dimensional details of the calculational model are shown in Figures 8.3 and 8.4.

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8.1.5 Press Feed Hopper

The press feed hopper is filled at the end of the dryer belt, where the granulator is located and is on a scale while being filled. (See Figure 8.4). Zinc sterate (0.25 wt %) is added to the UO_2 powder in the granulator. When the proper weight is attained the system is shutdown and the press feed hopper fill valve closed. This valve will remain closed until it is hooked up to the press feed line. The hopper is taken to the press feed roller station on a roller lift. From the roller station it is taken by the lift to the elevator, which elevates one press feed hopper at a time to the press feed mezzanine. The roller lift on the mezzanine places the press feed hopper in a press feed position or a press feed hopper storage location. The minimum center to center distance between press feed hopper in either the press feed or storage position is 5.0 ft.

Criticality Safety Analysis

Although only one press feed hopper is permitted in any designated area, an analysis was performed in order to account for the possibility of one press feed hopper being immediately adjacent to another press feed hopper. The following conservative assumptions were incorporated into the calculational model:

1. Two large hoppers (11 inch O.D. x 40 inch long cylinders) were assumed filled to capacity with UO2 powder with density of 3.5 g/cc at an enrichment of 5.0 wt % U235 and a water content of 5.0 wt %, which is twice the water content after drying on the belt dryer.

2. Both hoppers were assumed to be covered with a 0.25" film of water.
3. An external mist of 0.001 g/cc was assumed.
4. The concrete floor and ceiling have been accounted for in the models.

Sixteen group Hansen-Roach cross sections were used in KENO-IV to determine the reactivity of the system. Based on the conditions described above, a Keff = 0.3633 $\pm$ .0042 was obtained.

Although it is very unlikely that a filled press feed hopper could be fully reflected due to the locations of the storage press feed stations and configuration of the lifts, an analysis was done for a single press feed hopper completely filled, with 5.0 wt% UO_2 with a water content of 5.0 wt%. The resulting Keff = 0.6641 ± 0.0077.

8.1.6 Final Mixing

Filled press feed hoppers may be rolled to assure complete blending of the die lubricant. The press feed hopper roller is limited to one press feed hopper and the roller is surrounded by a wire mesh enclosure. The calculated Keff is less than the Keff for the two touching press feed hopper stated in Section 8.1.5.

8.1.7 Pressing

The filled portable hoppers are transferred to the pelletizing presses and secured to assure their stability and the containment of powder. Powder is gravity fed to the press, and compacted to green pellets which are placed into furnace boats. The pellets in the boats have to meet the slab height limit in Table 4.2.5. Only one boat shall be at each press at any one time. Each press is provided a spacing area of at least 20 square feet. The press is provided with enclosures which assure adequate ventilation at the opening face, and at the junction of the portable hopper with the press. Air flow rates are sufficient to assure face velocities of at least 100 fpm.

8.1.8 Dewaxing and Sintering

Furnace boats containing green pellets are charged in a single line to a dewaxing furnace, and then to a sintering furnace, where under controlled conditions, the pellets attain the desired properties. Because the UO2 is in a compacted form, dusting is minimal. Hydrogen burn off exhaust is vented from the building, and is filtered and monitored as specified in section 3.2.3 of Part I. The furnaces and their interconnecting conveyors are slab limited, with pellet slab heights as specified in Table 4.2.5. Stored furnace boats containing sintered pellets are limited to a maximum slab thickness specified in Table 4.2.5.

8.1.9 Final Sizing

Sintered pellets are transferred to the grinder feed system where they are aligned for the grinding operation which is carried out under a stream of coolant. The coolant is centrifuged to remove solids, and is recirculated at a uranium concentration considerably less than one gm/l. The infeeder, grinder and outfeeder have pellet configurations limited to a slab thicknesses as specified in Table 4.2.5.

Grinder sludge is removed from the centrifuge and dried in an oven. The dried material is subsequently stored in the concrete block storage area awaiting final disposition. An enclosure is provided around the grinder to preclude the dusting of UO2. The enclosure is maintained at a negative pressure with respect to the room.

The grinder coolant may collect in a one inch deep sump in the grinder and in a 24 liter sump behind the grinder, as shown in Figure B-1. A 24 liter sump is a factor of 1.3 less than a fully reflected optimally moderated critical volume, which is 31.0 liters as shown in Figure I.D.10 of UKAEA AHSB Handbook 1. Experience has shown that no appreciable sludge accumulates in the grinder sump. The centrifuge has a volume of 23 liters. The centrifuge is located in a closed hood. The critical volume for an unreflected volume of optimally moderated 5.0 wt% U-235 is 56 liters. The critical volume for a water reflected optimally moderated 5.0 wt% U-235 is 31.0 liters. These results were taken from Figures I.D.10 and I.D.14 of UKAEA AHSB Handbook 1. Applying a safety factor of 1.3 results in a safe volume of 43 liters for an unreflected volume and 24 liters for a reflected volume. Since the centrifuge has a steel wall and a steel internal structure which provides internal neutro absorption, the centrifuge is considered a safe unit. The centrifuge is cleaned periodically as required to permit continued operation.

Nevertheless, Figure B-1 does show spacing for the grinder sump to allow for any UO2 settling which may occur. Grinder coolant is normally recirculated, but may be disposed of by evaporation, or by discharge to the radiation waste system. Pellets are transferred by hand in a 2 inch high covered pellet tray to a storage rack or to a low temperature bulk drying furnace where any trace amounts of moisture are removed prior to rod loading. Both are limited to a slab thickness of 4.0".

8.2 Scrap Recycle

All clean scrap is accumulated for reprocessing and recycle with the feed material. Scrap may be milled to yield desired particle size best suited for the processing, oxidized and reduced to assure removal of volatile additives and to achieve the desired ceramic properties of the resulting recycle UO2, and blended to assure uniformity. The following equipment is included in the pellet shop annex:

a) Oxidation and reduction furnace

b) Milling equipment

c) General purpose

d) Filter Knockdown Hood

e) Blender

f) Micronizer

a) Oxidation and Reduction Furnace

This furnace is made up of two individual sections connected together. Product moves through both furnaces on a wire mesh belt. The oxidation section is used to heat sintered scrap in air to convert the UO_2 to U_3O_8. The reduction section is used to convert U_3O_8 to UO_2 in a heated gas atmosphere of H_2 and N_2.

b) Milling Equipment

This is a mechanical impact type grinder which uses a rotor blade assembly to grind UO_2 powder to a finer particle size so that the UO_2 powder can be recycled back into the pellet process. The UO_2 powder is fed through the top of the mill, passes into the milling chamber, passes through a screen and into a pail connected to the discharge of the mill. Connected to the discharge section is a vacuum cleaner to prevent the air pressure build-up in the milling during the chamber operation.

c) <u>General Purpose Hood</u>

This is a ventilated hood which is used for miscellaneous work involving handling of UO_2 powder or UO_2 contaminated material.

d) <u>Filter Knockdown Hood</u>

This is a ventilated glove box hood which is used to remove loose UO_2 powder from used absolute filters and prefilters.

e) <u>Blender</u>

The blender houses a sealed pail which contains UO_2 powder. The sealed pail is tumbled by rotating the pail in a non-concentric rotating motion. This action mixed recycle powder into a homogenous type mixture for use in the production pellet line.

f) <u>Micronizer</u>

This is an air impacting type of grinder. Fine particles of UO_2 powder are fed into the micronizer grinding chamber using a vibratory type feeder. High pressure air is then introduced into the grinding chamber. This action causes UO_2 particles to impact other UO_2 particles at high velocity resulting in finer particles. The powder fines and air mixture enters a bag house (sock filters) where the UO_2 particles are separated from the air. The UO_2 powder is collected in a pail while the air is exhausted into the FA-4 HEPA filter system.

The criticality safety for the furnace is based upon the slab limit as specified in Table 4.2.5. The remaining operations except blending, are all carried out in hoods with sufficient ventilation to assure a face velocity of 100 fpm. These operations are controlled by use of a 35.0 kg mass limit in accordance with Table 4.2.5 with spacing provisions taken from Table 4.2.6 of Part I of the application as shown in Figure 8-1. Positive spacing fixtures are used to assure spacing wherever more than one SIU is allowed in any given hood or box. A material balance log is maintained at the Milling Hood and Micronizer to provide additional assurance that the criticality limit of a 35.0 kg UO_2 mass limit will not be exceeded at these locations.

8.3 Storage and Transfer

8.3.1 Concrete Block Storage Area

A concrete block storage area is provided as shown in Figure 8-1. This storage area is intended for 5 gallon pails containing a maximum of 35.0 kg of UO2 and has a maximum height of 7 feet. The blocks are of solid 10" thick concrete, having a minimum density of 125 lb/ft3. Mortar is used to join the blocks and to secure the structure to the building wall. Steel shelves, of at least 16 ga. thickness are built into the structure with a vertical spacing of at least 16 inches. Each storage position measures 16" wide x 14" deep, and is lined on three sides with 1/4" thick mild steel. The criticality safety analysis demonstrates that the spacing boundary can be located 48 inches from the front of the shelves.

Criticality Safety Analysis

The following conservative assumptions were incorporated in the calculational model of the Concrete Block Storage Areas:

1. All steel structural materials were neglected.
2. An external mist of .001 g/cc was assumed.
3. Each storage position was assumed to be full with a 5-gallon steel bucket containing a homogeneous mixture of 35 kg UO2 at 5.0 wt % U235 and 5 wt % H2O. This mixture was assumed to be uniformly distributed within the bucket.
4. A 0.25" film of water has been assumed on the exterior steel walls of the shelving, the top of the shelves, and the exterior of each bucket.

The KENO-IV code with sixteen group Hansen-Roach cross sections was used to determine reactivity of the Concrete Block Storage Areas under the conditions noted above. A Keff of 0.3104 ± .0049 was obtained for an infinite system in the horizontal direction. The dimensional details of the calculational model are shown in Figure 8.5.

Another analysis was done for the Concrete Block Storage Area which includes a 12" water reflector in front of the storage area. The resulting Keff is 0.4698 ± 0.0070. A further analysis was done assuming the same bucket was completely flooded with water and reflected in front with 12" of water. The resulting Keff is 0.9221 ± 0.0070. This analysis did not include the steel structures, which includes the 1/4" steel on each side and the 1/4" steel shelves.

8.3.2 Pellet Storage Shelves

Steel shelves are provided for pellet storage. The shelves are 3 high. They have a width of 30" and are limited to a slab thickness of 4.0". The slab thickness of 4.0" is assured by limiting the number of fuel pellet trays stacked at any position. The entire storage array is covered by a sheet metal top which would prevent significant moderation of the array from discharge of the overhead sprinkler system. Water firefighting is not permitted in the pellet shop. Even though the pellet trays are normally covered and the storage shelves are covered, it was assumed in the analysis that the pellet trays were filled with water.

Criticality Safety Analyses

The following conservative assumptions were incorporated in the calculational model of the Pellet Storage Area:

1. All steel structural materials were neglected.
2. An external mist of .001 g/cc was assumed.
3. Each of the three storage shelves were assumed to hold a 4" thickness of a homogeneous mixture of UO2 at 5 wt % U235 at a density of 5.686 g/cc based on the random loading of pellets and water at maximum moderation (volume-weighted based on a UO2 density of 10.96 g/cc).
4. A 0.25" film of water has been assumed on the top shelf, which is empty and the back wall of the shelving.
5. The concrete ceiling (4"), floor (16") and the back wall (8" concrete block equivalent to 5" solid concrete) were also included.

The KENO-IV code with sixteen group Hansen-Roach cross sections was used to determine reactivity of the Pellet Storage Area under the conditions noted above. A Keff of 0.7468 ± .0049 was obtained for an infinite system in the horizontal direction.

The UO2 loading of the trays was determined by doing a total of 14 mesurements. The pellets pack to an average density of 5.95 gm UO2 per cc (5.24 gm U per cc, with a 2 sigma variation of 0.264).

The 16 group cross sections for the pellets were calculated for 0.3766" diameter pellets. Dimensional details of the calculational model is shown in Figure 8.6.

8.3.3 Rod Transfer

Flat carts measuring 3' x 13'-1/2" are used for transporting up to two steel boxes with inside dimensions of 5-1/2" x 8" x 14'4" long, each containing over 300 fuel rods. The rods are assumed to be in a close packed hexagonal lattice with a maximum water to UO2 volume ratio of 0.48, based on a rod O.D. of 0.44" and a pellet O.D. of 0.3765".

From Figure 1.E.16 of UKAEA Handbook AHSB 1, the critical infinite slab thickness for 5.0% enrichment fully reflected is about 8 inches for this degree of moderation. Applying the safety factor of 1.2 yields an allowable slab thickness of about 6.7 inches. Accordingly, the rod transfer cart with two 5.5 inches deep boxes is safe as long as the rods are not stacked higher than 6 inches in each box. Carts may be placed alongside each other, or will be spaced a minimum of 1 foot from other fissile material.

8.3.4 Transfer of Material

Material may be transferred on carts which accommodate one mass or slab limited SIU, or may be transferred by hand, one SIU at a time. Carts used for mass limited SIU's shall provide for centering of the unit, and shall measure at least 2.6 feet on a side as specified in Table 4.2.6. Because most spacing areas do not extend beyond the physical boundary of the equipment, spacing between transfer carts and the equipment is of no concern. In cases where the spacing area entends beyond the equipment boundaries, such extends beyond the equipment boundaries, such as the storage facilities, the spacing boundary will be indicated by a colored line. The line may be crossed by carts only when they contain no more than one mass or slab limited SIU, and then only to permit an operator to transfer that SIU to an available storage position.

8.4 Pre-treatment of Low Level Liquid Wastes

Aqueous wastes from low level radioactive cleanup operations such as mop buckets and decontamination solutions are processed through a system designed to remove particulate matter. This system consists of a prefilter and dump tank, a high efficiency (double concentric bowl) centrifuge, a slant storage tank, an open faced ventilated hood, a drying oven, and sundry valves, piping and pumps. Figure 8.14 shows a sketch of the liquid waste processing system layout in the Building 17 Annex. Access to the open face ventilated hood is on the long side facing away from the slant storage tank. The 96 inch long hood has three regions. On the right and left hand sides are work surfaces about 42 inches above the floor. The left hand work surface is about 22 inches in width and 42 inches in depth; this surface is used in the dismantling and cleaning of the centrifuge. The right hand surface is approximately 40 inches in width nd 54 inches in depth over the majority of the width. A removable drying oven, approximately 30 x 27 inches, is normally located at the back of this work surface. The front part of the work surface contains a covered powder funnel attached to a 6 inch diameter flexible hose which is inturn coupled to a five gallon bucket sitting at floor level in a fully enclosed bottom section of the region of the hood. An access door on the face side of the hood permits access to the bucket. The central region of the hood contains the Westfalia Clarifier (centrifuge) bolted to a pad on the concrete floor.

The clarifier bowls are below the 42 inch high working levels of the adjacent sections of this hood.

Contaminated liquids are batch processed by this system until sampling checks verify that the activity level is below a specific threshold prior to pumping the contents of the slant (storage) tank to the Building 6 liquid holding tanks. Solids removed from the prefilter and centrifuge are handled and processed as dirty residue (contaminated scrap) material. The overall process is depicted in the flow chart of Figure 8.15. A more detailed description of the process and equipment follows.

The principal components of the liquid processing part of the system are as follows.

1) Prefilter and Dump Tank - The prefilter is located at the inlet to the dump tank and consists of a 20 mesh (0.034") screen backed up by a coarser mesh screen for mechanical support. The dump tank has a capacity of 5 gallons, and is raised above the floor far enough (~9 inches) to provide gravity feed to the dump tank pump.

2) Dump Tank Pump - This pump is located at floor level and is used to pump the contents of the dump tank into the centrifuge. This pump is non-reversible. A check valve is in the outlet line to prevent back flow to the dump tank.

3) Centrifuge - The centrifuge is a Westfalia Clarifier manufactured by Westfalia Separator AG. It is a high efficiency, twin (concentric) bowl system having a total capacity of 19 liters. The bowls are concentric as illustrated in Figure 8.16.

4) Centrifuge Hood - The centrifuge hood is an open face hood which provides forced ventilation to the drying oven, centrifuge, and centrifuge cleaning operation. This hood also minimizes the amount of water incident upon the centrifuge and peripheral equipment within the hood from water emanating from the fire sprinkler system.

5) Slant Tank - The slant tank is a slab geometry stainless steel (304) storage tank. The outer length and breadth are approximately 48 x 54 inches, the internal thickness is ≤4.0 inches, and the wall thicknesses are 11 guage (0.125"). The tank has five internal welded angle struts to preserve the thickness dimension. One strut is at the center and two along each main diagonal, each of the latter being two thirds of the way from the corner to the center strut. Access ports are provided in one of the major surfaces for inspection. Fittings are provided for inlets, outlet, vent line, and a sight glass line. The tank is in a near horizontal plane; two top diagonally opposite

corners are at an elevation of approximately 49 inches and the remaining two are at 52 and 46 inches. The outlet is in the bottom face at the bottom most corner. The inlet and vent fittings are in the upper surface near the highest corner. The sight glass fitting is near the exit fitting. The tank employs 1-1/2" x 1-1/2" x 1/4" angle stiffeners along the diagonals beneath the lower surface. In addition to the four corner legs and bracing, a center support leg is employed to supplement the diagonal stiffeners.

6) Overflow Tank - The overflow tank is on the floor, made of stainless steel, and has a capacity of $\leq$5 gallons. The large diameter overflow line exiting the slant tank and going to the overflow tank serves both as a vent and overflow to the slant tank. The overflow tank also receives overflow liquid from the centrifuge via the drain line from the lower housing of the centrifuge and a drain line from the floor of the center region of the hood that surrounds the upper part of the centrifuge.

7) Circulating Pump - The circulating pump is employed to recirculate the liquid from the slant tank to the centrifuge. Flow from the centrifuge to the slant tank occurs as a result of the pumping action of the operating centrifuge. This pump is non-reversible. The check valve in the outlet line of the circulating pump prevents backflow through this pump.

8) Discharge Pump - The discharge pump is employed to transfer the contents of the slant tank to the pipe line going to the Building 6 waste tanks.

The above components are plumbed and in the manner illustrated in Figure 8.17. Typical system operation is as follows; here it is assumed all pumps are initially off:

The waste liquid is poured through the prefilter to the dump tank. When the dump tank pump is turned on, the contents of the dump tank are transferred to the centrifuge. The centrifuge is normally operating: if the centrifuge is not turned on and the bowls are full, the water wi[illegible] spill through the overflow rather than go to the slant tank. Pumpin[illegible] action of the operating centrifuge is required for the excess liquid flowing to the centrifuge to be directed to the slant tank. Thus, all contaminated liquid flows through the operating centrifuge prior to filling the slant tank. Note that as long as the dump tank pump is operating, the discharge pump cannot be turned on. When the dump tank is empty, the dump tank pump is turned off.

To recirculate the contents of the slant tank through the centrifuge, the circulating pump is turned on. Recirculation through the operating centrifuge continues until the clarity of the water flowing through a sight glass in the centrifuge discharge line is judged to be acceptable. If clarity does not improve sufficiently, the centrifuge may require cleaning. If clarity is acceptable, a sample is withdrawn, dried, and counted to determine residual contamination. If not acceptable, further recirculation is carried out. If acceptable, the circulating pump is turned off and valve 7 is locked into the off position. A sample of the contents of the storage tank is extracted at a point below the lowest level of the tank. This sample is dried and counted; if acceptable, the contents of the tank are ready to be diverted to the drain line going to the holding tanks at Building 6. During the time interval between extraction of the sample from the storage tank and the decision to dump the contents of the tank to Building 6, the contents of the slant tank are protected from change by the locked valve (7). Should additional liquid be injected into the centrifuge system through the dump tank or the wash basin, this liquid would be ejected to the overflow tank.

The contents of the storage tank are transferred to the holding tanks at Building 6 by first turning on the discharge pump. This automatically closes the second barrier against additional waste water entering the storage tank from the centrifuge and opens the solenoid valve between the discharge pump and valve 8. Upon opening valve 8, the contents of the storage tank are pumped to the holding tanks of Building 6. When the slant tank is emptied, the discharge pump is turned off and the valves reset to refill the slant tank, if more liquid is available to process. If the centrifuge requires cleaning, it is turned off, dismantled, and the solids are cleaned out of the two bowls and collected in a pan and dried in the oven.

If the second sample is not acceptable, further centrifuging of the contents of the storage tank is required. To revert to a recirculation mode between the slant storage tank and centrifuge, valve 7 must be unlocked and the circulating pump turned on.

The working floor level in the vicinity of the dump and overflow tanks is raised about one inch above the concrete floor by a steel grating. This grating is contained within a large steel pan approximately 3 inches in depth so as to contain liquid spilled in handling operations.

Nuclear Safety

Nuclear safety of the liquid waste processing system is predicated on the following observations and conclusions. It will be noted that the primary barrier against criticality is the use of geometrically favorable containers. Secondary barriers consist of engineered design features and administrative controls.

1) The prefilter in the dump tank screens out particles larger than 0.034 inches from entering the dump tank.

2) The dump tank has a capacity of 5 gallons or 18.9 liters. This value is 26% less than the critical, fully reflected volume of 25.5 liters

inferred from the most conservative data of Figure 8.18 for 0.050 inch diameter pellets. In the event that the prefilter failed, the dump tank is still 15% less than the critical, fully reflected volume of 22.1 liters for the optimum pellet diameter of 0.3 inch diameter pellets. The bottom of the dump tank is about nine inches off the floor, consequently the likelihood of full reflection is small as is the likelihood of having the 0.3 inch diameter pellets uniformly distributed throughout the volume of the dump tank with a water to oxide volume ratio of about 2.8.

3) All solution pumped into the slant tank has passed through the operating centrifuge. Therefore, the larger particles should be removed from the solution entering the slant tank providing the sludge regions of the centrifuge bowls are not fully loaded. The slant tank has a slab geometry with a maximum solution thickness of $\leq$4.0 inches (see discussion below). In addition, the slant tank has a screen barrier around it so as to prevent the close approach of any significant moderating type material to either of the major faces of the slab.

Under normal operating conditions, the concentration of UO2 in the slant tank is sufficiently low that it is impossible to achieve criticality regardless of the tank volume or geometry. The concentration of UO2 in the slant tank only approaches that of the solution being poured into the dump tank when the sludge volume of the centrifuge bowls approaches full capacity. Nevertheless, the slant tank geometry is set so as to preclude criticality in the event that the slant tank is fully reflected and filled with a uniform distribution of optimally sized UO2 pellets at optimum moderation.

The most conservative data of Figure 8.18 on critical slab thickness for an optimally moderated and reflected slab versus particle size shows that the minimum slab thickness occurs for particles/pellets having a diameter of 0.2 to 0.4 inches. The critical slab thickness is 4.15 inches. The corresponding water to oxide volume ratio is about 2.3. If the presence of the screen barrier is assumed to reduce the reflection of the tank by 50%, this is equivalent to an increase in the critical slab thickness of 1.6 inches for optimum moderation conditions within the tank. The slant tank is approximately four feet off the floor, consequently flooding of the surrounding area so as to reflect the tank is highly improbable. An approach to criticality, even under the postulated failure of the prefilter and centrifuge to remove UO2 particles from the solution entering the slant tank, cannot occur as long as the slant tank thickness is maintained.

The slant tank engineering design is such that dimensional changes with postulated loading of the tank are minimized. In the absence of any structural supports other than at the edge of the tank and calculating the deflections for two coupled (via five internal braces)

1/8 inch thick plates, it was estimated that the deflection resulting from a mass distribution of liquid of density 1 g/cc (pure water) would be 0.128 inches; for a contaminated solution density of 2.5 g/cc, the deflection is estimated at 0.249 inches; and for a solution density of 3.5 g/cc, the deflection is 0.333 inches. To minimize deflections, the 1.5 x 1.5 x 0.25 angle braces were run diagonally along the lower face of the tank and a central support leg to the floor was added.

In summary, the slant tank is structurally reinforced and vented so as to minimize possible deflection of the tank and enlargement of the liquid slab thickness. The critical slab thickness for optimum moderation and particle size conditions within the slant tank and assuming half reflection of the tank is conservatively estimated as 5.75 inches. This value is based on using the most adverse data of Figure 8.18 as well as the critical buckling and reflector savings data of DP-1014 for 5 w/o enriched UO2. This derived value exceeds the 4.0 inch maximum design thickness criterion for the slant tank by 44 percent.

4) The centrifuge has twin, concentric bowls with a total capacity of 19 liters. This volume is sufficiently close to that of the dump tank (18.9 liters) that the same nuclear safety arguments of item 2, above, apply.

5) As noted in the descriptive section, the hood is a three part hood. The central section is occupied by the centrifuge. The right and left sections are designated as mass limited regions; the contaminated scrap is handled under the SIU mass limits defined in Chapter 4 of Part I. The central region of the hood containing the centrifuge has a floor that is below the right and left work surfaces by about 20 inches. However, this well type area is drained by a line going to the overflow tank. It is also noted that a city water line enters the hood but the valve is exterior to the hood, thus, should the line break within the hood, it would not flood the hood.

6) The overflow tank is a five gallon, or less, capacity stainless vessel. As noted above in the discussion of item 2 (and 4) above, all scenarios involving 5 gallon or 19 liter containers are safe.

7) As noted in Figure 8.14, the centrifuge complex is located in the Annex near the stairway leading to the mezzanine. Since waste processing is planned for the mezzanine area, one pathway for waste is up the stairway over the slant tank. Therefore, scenarios of possible interest have to do with potential neutronic interaction between media on the stairway and material in the slant tank.

The stairway employs an open grill type of stair tread, thus material can fall through it. However, a barrier in the form of sheet metal has been attached to the beams supporting the stair tread and rail. Should material be spilled on the stairs and pass through the open treads, it will strike the sheet metal and slide downward away from the slant tank. One scenario of interest postulates that a 35 Kg amount of 5 w/o enriched UO2 powder is being carried up the stairway and the 5 gallon container also has sufficient water in it to yield an optimum solution density (1.6 gU/cc) in the bottom of the five gallon container. The container is set on the stairway at the closest point of approach to the slant tank. For the geometry of Figure 8.14, the closest distance of approach is along the upper edge of tank closest to the stair tread. The minimum separation distance between the bucket and slant tank is calculated to be approximately 8 inches; the minimum separation distance between the sheet metal dust cover and the slant tank is approximately 3.5 inches.

To quantify the magnitude of the neutronic interaction between the postulated 5 gallon container of UO2 and the slant tank, the following conservative representation was modelled in a KENO-IV calculation. The slant tank was modelled as horizontal, four feet above a 16 inch thick concrete floor, and filled with a homogeneous mixture of 5 w/o enriched UO2 and water at optimum moderation. The tank internal dimensions were taken as 48" x 54" x 4" and the walls were taken as one eighth inch thick stainless steel. Twelve-inch thick vertical water walls were assumed along the four sides of the slab tank extending from the floor to a 20 foot level. At the latter level a twelve-inch water slab was modelled. The five gallon bucket was assumed to be 11.75" O.D., 13.25" tall, and having a 28 gauge steel wall thickness. The base of the bucket was five inches above the slab tank and centered on the face of the tank. The homogeneous mixture of optimally moderated UO2 was 10.909 inches deep in the bucket. The KENO-IV computed multiplication factor, using Hansen-Roach cross sections was 0.76086 $\pm$ 0.00489.

Additional analyses were done under the assumption that the UO2 in the bucket and the slant tank is heterogeneous material having an average particle diameter of 0.325 inches. The water to oxide volume ratio in both containers was taken as 2.4 which is close to optimum for the slab geometry slant tank. The UO2 - water depth in the bucket was taken to be the same as in the previous homogeneous UO2 - water calculation. By preserving the volume of the solution in the bucket the mass of UO2 increased in the heterogeneous calculation from 35 Kg UO2 to 58.4 Kg UO2. The calculation was run versus separation distance between the bucket and slant tank. Sixteen group heterogeneous cross sections were generated by the NITAWL-XSDRNPM routines and employed in the KENO-IV code to yield the following multiplication factors versus separation distance.

Separation Distance (inches)	Keff
2	0.81741 ± 0.00488
5	0.78378 ± 0.00462

From these homogeneous and heterogeneous calculations it is concluded that interactions between material passing up the stairway, resting upon the stairway, or spilled upon the dust cover attached to the under side of the stairway result in acceptable calculated subcriticality margins.

In view of: 1) the highly conservative interactive geometry assumed between the container and slant tank, 2) postulated loadings of the mass limited container (>150 lbs of H_2O/UO_2) and slant tank (optimum moderation and particle size) and 3) complete reflection of array, the above analysis constitutes a worst case scenario.

8.5 Rod Loading and Assembly Fabrication

8.5.1 Pellet Stacking

Pellets from the pellet fabrication facility, or from outside vendors are placed on a table where they are aligned for rod loading. On the table, the pellet configuration is limited to the slab limit as specified in Table 4.2.5. The UO2 pellets are placed on troughs one pllet high before being loaded into rods.

8.5.2 Rod Loading and Fuel Rod Transport Carts

Pellets are transferred from stacking troughs into rods. The loaded rods are placed into carts each of which can hold up to 250 fuel rods in parallel sleeves which are spaced on four rings in an annular fixture with an I.D. of approximately 10 inches and an O.D. of approximately 22 inches. Guard rails prevent the carts from coming any closer than 3 feet center-to-center. The carts are used in normally dry areas to transfer the rods to operations which include end plug welding, weld deflashing and leak testing. The welding and deflashing operations are performed on one rod at a time. The leak testing operation is performed on two rods at a time. Welded and deflashed rods are immediately returned to the cart after each step is completed. Finished rods are fluoroscoped and are checked for enrichment with a maximum slab limit as specified in Table 4.2.5.

Criticality Safety Analysis

The following conservative assumptions are incorporated into the calcul   al model of the Rod Loading and Fuel Rod Transport Carts:

1) the 1/4 inch thick, 8" O.D. inner steel cylindrical annulus was accounted for in the model. All other construction material was neglected.

2) The carts were assumed to be infinite array in the x and z directions.

3) A mist of .001 g/cc water was assumed for all air spaces.

4) The fuel rods are contained in 1/2 inch, Sch 40 PVC tubes, each 134 inches long. There are 250 tubes arranged in 4 concentric rings with an average pitch of 1.303 inches. The fuel tube region of the cart is thus a cylindrical annulus beginning at 7.445" from the centerline of the cart and extending to a radius of 12.711 inches. On either side of the fuel tube region, is a weld sample box (4.375" x 4.375") attached to the inner side of the cart. The weld sample boxes contain a 5x5 array of the PVC tubes which hold empty fuel rods for the purposes of weld sampling only. A cover of 1/4 inch aluminum with plexiglass areas encloses the top, sides, and back of the cart.

   In the calculational model it has been assumed that all 250 positions in the cylindrical area and all 50 positions in the weld sample boxes were occupied by the largest diameter rods (0.3765" O.D. UO_2 pellets at 10.061 gm/cc stacked density with a Zr-4 cladding thickness of .028 inch) at the maximum enrichment of 5.0 wt % H_2O. It was also assumed that the fuel rods and the

PVC tubes extended the full length of the cart (165 inches). The moderation effects of the PVC have been included in the analysis, however the absorption effects have been neglected. A 0.25" film of water has been assumed on the exterior sides of the cover. The concrete floor and ceiling have also been modelled.

The NITAWL and XSDRNPM codes were used to obtain 16-group cross sections from the 123-group GAM-THERMOS library for input to XENO-IV. A reactivity for the Rod Loading and Fuel Rod Transfer Cart, keff = 0.873 ± .0058, was obtained based on the conditions described. Dimensional details of the calculational model are shown in Figure 8.7.

8.5.3 Autoclave Corrosion Test

Two autoclaves used for corrosion testing of finished fuel rods are shown in Figure 8-1. The stainless steel tanks are 14 feet long and have an inside diameter of 14 inches with wall thickness of 1.5 inches. The center line distance between autoclaves is a minimum of 66 inches. Each autoclave is limited to 32 fuel rods by administrative control. The fuel rods are held by stainless steel fixtures consisting of eight plates which are five inches wide and 1/8 inch thick. During operation, the interior of the autoclave could conceivably experience all conditions of water moderation, from completely dry to full density water. Criticality safety of the autoclaves is based on dimensional comparison with the fuel assembly storage area. The fuel assemblies have been designed for maximum reactivity and have a keff of less than 0.90 in full density water. (See section 8.5.7). The rod spacing in the fuel assembly is thus the optimum. If the fuel rods were aligned in the autoclave at this optimum spacing, it would thus take 256 rods to achieve a keff of approximately 0.90. The maximum number of rods allowed (32) provides a large margin of safety under all conditions of moderation and reflection. Even with all autoclaves filled, the number of fuel rods present (192) would be less than the number required for one fuel assembly of the 16 x 16 type.

8.5.4 Fuel Rod Storage Area

The multi-level storage area shown in Figure 8-8 for boxes of fuel rods consists of up to 10 tiers of 32 locations each. The steel fuel rod boxes have a maximum length of 14'-4" and an inside width and depth of 8 inches and 5-3/8 inches, respectively. A vertical spacing of 12-1/2 inches between boxes is maintained, the first tier being 18 inches above the concrete floor. Lateral spacing is restricted by physical barriers to a minimum of 4 inches. The rod boxes rest on roller conveyers to facilitate movement in and out of the storage array and are held in place by a fixed brace. The entire storage array is covered by a sheet metal roof to assure the exclusion of sprinkler water. The fire resistant roof has 3% pitch to assure adequate drainage to the floor. Water accumulation in the vicinity of the storage rack is not considered credible in view of the close proximity of an open equipment pit in the floor which is 30 feet x 60 feet x 18 feet deep. A 3 foot deep sump at the bottom of the pit is equipped with a level detector which activates a pump to transfer any accumulated water to the industrial sewer system.

Criticality Safety Analysis

The following conservative assumptions were incorporated in the calculational model of the Fuel Rod Storage Area (Figure 8.8):

1) Each of the rod boxes was assumed to contain the smallest diameter fuel rods (0.382") at an enrichment of 5.0 wt % U235 with a density of 10.061 g/cc. The fuel rods were assumed to be tightly packed in an hexagonal array. The 8" wide box was filled to a height of 6.25" containing 371 fuel rods, which is greater than the slab limit (6 inches) as specified in Table 4.2.5. The fuel was assumed to be dry. The fuel and clad were homogenized over the volume of the box.
2) A vertical spacing of 11.5" between rod boxes was assumed.

3) A lateral separation distance of 3.5 inches between rod boxes was assumed. Interspersed moderation was not considered credible since moderation control is assured by the cover, walls, and doors and the storage area.

4) All steel construction material was neglected.

5) The concrete ceiling (4") and floor (16") have been included in the calculation.

6) The rubber pad was modeled as water in the position in the box as shown in Figure 8.8.

The NITAWL and XSDRNPM codes were used to obtain 16-group cross sections from the 123-group GAM-THERMOS library for input to KENO-IV, the code which was used to determine reactivity of the Fuel Rod Storage Area under the conditions noted above. A keff = 0.6850 ± .0032, was obtained for a system with four tiers in the vertical direction and infinite array of boxes in the horizontal direction.

8.5.5 Double Shelf Rod Storage Rack

The double shelf storage racks for fuel rods hold a maximum of 12 steel boxes identical in all respects to those in the multi-tier array described above. Each box is equipped with a tight fitting Aluminum cover which overlaps the outside edge of the box by a minimum of one inch. One box may remain uncovered for short periods of time to allow for the addition or removal of rods for inspection purposes provided that personnel are in attendance. Spacing between boxes in both a vertical and horizontal direction is a minimum of 6 inches. Minimum center-to-center spacing between storage racks is 55 inches and the racks are considered to be present in an infinite array in the horizontal plane. The location of these racks is shown as in Figure 8-1.

Criticality Safety Analyses

The following conservative assumptions were incorporated in the calculational model of the Double Shelf Rod Storage Racks (Figure 8.9):

1. Each of the steel boxes was assumed to contain the smallest diameter fuel rods (0.382") at an enrichment of 5.0 wt % U235 with density of 10.061 g/cc. The fuel rods were assumed to be tightly packed in an hexagonal array. The 8" wide box was filled to a height of 6.25" containing 371 fuel rods. The fuel was assumed to be dry. The fuel and clad were homogenized over the volume of the box.
2. The 0.125" pad at the bottom of the box was modelled as water.
3. All structural materials were neglected.
4. An external mist of 0.001 g/cc was assumed.

5. A 0.25" film of water has been assumed on the cover and sides of the box and on the supports on either side of the box, which were conservatively assumed to extend the full length of the box.

6. The concrete ceiling (4") and floor (18") have been included in the calculation.

The NITAWL and XSDRNPM codes were used to obtain 16-group cross sections from the 123-group GAM-THERMOS library for input to KENO-IV, the code which was used to determine reactivity of the Double Shelf Rod Storage Racks under the conditions noted above. A keff = 0.9144 ± .0065, was obtained for an infinite system in the horizontal plane.

8.5.6 <u>Fuel Assembly Fabrication</u>

Fuel rods are loaded into the assembly skeleton in a fixture which provides a lubricating water spray. These fixtures are designed to assure that water cannot be retained. The Keff for an isolated assembly is less than 0.90. See Figure 8.10 for all Dimensions of the Fuel assembly.

8.5.7 In-Plant Storage of Fuel Assemblies

Fuel assemblies are stored in a vertical position using racks of adequate strength to preclude loss of the design spacing. The assemblies in the storage positions only shall be wrapped with polyethylene with the bottom ends open to assure free drainage. There are 440 storage positions and an adjacent inspection area consisting of 16 positions. Within the same room, (but at greater separation distances) there are two horizontal loading tables where the fuel rods are initially loaded into the assembly skeletons, a vertical wash tank where the assemblies receive a final demineralized water rinse, two fixed vertical inspection stands equipped with elevator platforms to allow final Q.C. dimensional checks, and a marked floor area where the assemblies are loaded into shipping containers prior to outdoor storage. Each of these stations is physically limited to one fuel assembly except the shipping container which holds two. The assembly storage room can thus contain a maximum of 465 fuel assemblies, 440 storage positions, plus 25 additional locations. All assemblies outside of shipping containers shall be stored vertically within the design spacing criteria of the Assembly Storage Room shown on Figure 8.11.

1) A 20 x 34 array of assemblies was conservatively modeled at a 9.75 inch center-to-center spacing of fuel assemblies within the double rows. The actual average minimum center to center distance within the fuel storage racks is 10 inches. The distance between rows of fuel assemblies within any given double rack is 35 inches center-to-center while the aisle between the double racks is 37 inches (center-to-center). This calculational array effectively brings the 25 additional assemblies closer together and provides greater interaction with the 440 assemblies in the storage area than is actually possible. The calculational array thus contains 680 assemblies while the maximum number in the room is limited to 465. (See Appendix 8-1, drawing No. NFM-E-4229, "Criticality Model Fuel Assembly Storage Room." By squaring off the racks and totaling the number of fuel assemblies the number 680 is arrived at).
2) All steel construction material was neglected.
3) The water mist density has been calculated to be 0.000075 grams per cubic centimeter (see section 8.7). For conservatism a water-mist density of 0.001 grams per cubic centimeter was assumed to be in and around the fuel assemblies in the storage array. (This is a factor of about 13 times higher than the mist density calculated in section 8.7 or about 17 times higher than the mist density calculated in Appendix D for a single sprinkler head at maximum flow and pressure). A uniform

water film thickness of 0.025 centimeters was assumed on the fuel assembly surface. The actual calculated film thickness with a 15% uncertainty was 0.0094 centimeters (see Section 8.8). This calculated film thickness is for 50 degrees F water, while the minimum ambient temperature is actually higher.

4) One hundred twenty three (123) DLC-16 energy group cross sections were used to calculate the reactivity for an infinite fuel storage array using KENO. The 123 group was collapsed to 16 groups using XSDRNPM end the reactivity calculated for an infinite fuel storage array. The resultant reactivities for 4.1 wt % U235 Fuel were 1.00158 ± 0.00608 and 1.00074 ± 0.00569 for the 123 and 16 groups, respectively. Since the reactivities are essentially the same within the statistical uncertainty of KENO the 5.0 wt % U235 finite fuel storage array was done using 16 energy groups.

5) The 16 energy group cross sections were generated using XSDRNPM for the 8" concrete walls, 16 inch concrete floor, 4 inch concrete ceiling and the external water mist between the fuel assembly array and the ceilings and the walls. The 16 group cross section sets described above were then used in KENO-IV to determine the reactivity of the fuel assembly storage area under the above noted conditions for the most reactive assemblies (the 16 x 16 type with the grids being neglected). Dimensional details of the calculational model and the fuel results obtained are shown in Figures 8.10 and 8.11.

The resulting Keff for the finite fuel storage array is 0.842 ± 0.004 which is well below a Keff of 0.95. Using the same methodology additional cases were analyzed for a fuel enrichment of 4.1 wt % U235 where the fuel assembly center to center spacing and the water film thickness were varied to determine the effect on reactivity.

A tabulation of Assembly Spacing/Mist Density/Film Thickness Reactivity Values follows:

Assembly c/c Spacing	Mist Density	Film Thickness	Reactivity for a Finite Array
9.75"	0.001 gms/cc	0	0.69575 ± 0.00397
9.75"	0.001 gms/cc	0.0094 cm	0.732 ± 0.004 (see note 1)
9.75"	0.001 gms/cc	0.025 cm	0.77224 ± 0.00349
9.75"	0.001 gms/cc	0.055 cm	0.89932 ± 0.00341
10"	0.001 gm/cc	0	0.69913 ± 0.00422
10"	0.001 gms/cc	0.055 cm	0.904562 ± 0.00367

A validation of the methodology used to calculate the reactivity values noted is contained in Appendix C.

The local fire departments have been instructed to use only dry chemical extinguishing methods in the fuel assembly storage room

NOTE 1: This is an interpolated value

and the pellet shop. Signs restricting fire fighting in this area to dry chemical methods only have been posted at each entrance to the assembly storage room. There is only one vehicle access gate to the fuel fabrication facility. Thus, criticality safety is assured under all credible conditions of moderation.

8.5.8 <u>Shipping Container Storage</u>

Fuel bundle shipping containers (Models 927A1 and 927C1), each containing two fuel assemblies, are stored outdoors in arrays up to three high. The width and length will vary; thus, the quantity of containers is limited only by the width and length of the space allocated for storage. The steel shipping container, approximately 3 feet in diameter and up to 217" long, houses two fuel bundles of the types previously described in this license. The two bundles in each container are separated by six inches. An eight foot high chainlink fence encloses the storage area.

<u>Criticality Safety Analysis</u>

The following conservative assumptions were incorporated into the calculational model of the Shipping Container Storage Area:

1) The fuel assemblies are assumed to be made of 5.0 wt.% U235 enriched UO2 with no poison shims. The most reactive assemblies (the 16 x 16 design) were used.
2) The three high double infinite array of shipping container was analyzed.
3) The containers were assumed to be flooded and the array was reflected by 12" of water on the top and bottom of the array.

The NITAWL and XSDRNPM codes were used to obtain 16-group cross sections from the 123-group GAM-THERMOS library for input to KENO-IV, the code which was used to determine reactivity of the Shipping Container Storage Area under the conditions noted above. A keff = 0.9245 ± .0057 was obtained for the system peak reactivity of the system. The density of water within the exterior to the containers was made identical.

Details of the calculational model are shown in figure 8.12. This analysis also provides the basis for considering an open or closed assembly shipping container as an SIU which requires no spacing beyond the physical boundaries of the container. Accordingly, individual containers may be stored in the facility in unrestricted numbers.

8.5.9 Fuel Salvage

Off-specification fuel rods are received one rod at a time in a ventilated hood. The end cap is cut off and zirc chips are vacuumed from the rod. If the rod is not to be unloaded a temporary plug is installed in the rod before it is removed from the hood. If the rod is to be unloaded, the pellets are placed in a 2 inch high pellet grinder tray. The unloading operation is performed with ventilation being drawn across the tray.

This operation is considered a mass limited SIU, with limits taken from Table 4.2.5.

8.5.10 In-Process Storage of Fuel Pellets in Containers

Incoming drums of pellets shall be stored in their original containers only. Two containers are strapped to each pallet, one pallet high as required by the NRC certification of compliance. The pallets may be stored inside the Building 17/21 complex security fence or inside Building #21 within the limits of a Transport Index of 90. In Building #17 three pallets can be stored in the Pellet Shop Annex and four pallets can be stored in the Pellet Shop Rod Loading Area as discussed in Section 4.3.18. During storage in Building #17 the pallets can be stored next to each other but must be at least one foot from process equipment in the area. Arrays of different shipping containers shall be separated from each other by at least 20 feet.

8.5.11 Buckets containing 35.0kg UO_2

The UO_2 powder and UO_2 pellets may be stored in 5 gallons or less enclosed buckets. Normally the powder and pellets will be dry. The only time the buckets will be open will be in hoods, which will limit the amount of water that can be introduced in the bucket from the fire sprinklers.

Criticality Safety Analysis

A very conservative analysis was done for the following array of buckets filled with UO_2 powder. The conditions, assumptions, and results are as follows:

1) The steel cylindrical container has an effective inner diameter of 10.75" and an effective height of 14.25". A 2x2x2 array of containers was analyzed with the buckets in the array separated by 1 foot.

2) Each container was filled with 35.0 kg UO_2 enriched at 5.0 wt % U235 and water. The UO_2 was assumed to have a density of 10.96 g/cc. The water was assumed to fill the space not occupied by the UO_2 forming a mixture.

3) The KENO-IV code with sixteen group Hansen-Roach cross sections was used to determine the reactivity at the array of 8 containers. The K_{eff} for the full water density within the array is 0.91011 ± 0.01013. The K_{eff} for an external mist of 0.001 gm/cc is 0.76685 ± 0.00495.

4) To insure that the container with less than 35 kilograms of UO_2 was not more reactive an analysis was done as a function of UO_2 mass in the container. The reduction in the mass in the container results in a higher water to fuel ratio. The KENO results for a 2x2x2 array of containers are shown in Figure 8.13. It can be seen the maximum activity occurs at the 35.0 kilogram limit.

The following analysis was done for a bucket filled with 0.4" diameter pellets.

1) Sintered pellets, when randomly loaded pack to an average density of 5.95 gm/cc, with a one sigma variation of 0.264 as determined from a series of 14 measurements. Thus, at a 95% confidence level, the VH_2O/VUO_2 ratio does not exceed

   1.0 and from Fig 1 E.1 of UKAEA Handbook AHSB1, the critical mass for 5.0 wt % U235 at the $VH2O/VUO_2$ of 1.0 is in excess of 200 kg Uranium.

8.5.12 Slab Limits for Pellets

The following analysis was done for a slab filled with 0.4" diameter pellets.

Pellets, when randomly loaded, pack to an average density of 5.95 gm/cc, with a one sigma variation of 0.264, as determined from a series of 14 measurements. Thus, at a 95% confidence level, the Volume of H_2O to Volume at UO_2 ratio does not exceed 1.0 and from Fig. 1.E.16 of UKAEA Handbook ASHB1, the critical slab thickness is 6.2 inches. Dividing by the safety margin of 1.2 results in a slab thickness of 5.2 inches.

8.5.13 Fuel Rod Pre-Stacking Station

The Fuel Rod Pre-stacking Station is used to stack the fuel rods in the final array of fuel rods in the finished fuel assembly. Individual fuel rods are pulled from any one of the three trays into a rod box forming a close packed hexagonal array. As the position for the CEA guide tube is approached, a four rod cluster of empty tubes is placed in the array. The final array will contain five of these empty tube clusters at the CEA guide tube location. After the array is completed, the fuel rod box is loaded into the fuel rod box transporter, which is limited to one rod box and placed into the fuel rod storage area.

When a pre-stacked array is to be loaded into the fuel assembly grid within the fuel assembly area, the fuel rod storage box is moved to one of two possible storage positions where the fuel can be unloaded in the fuel assembly area. The fuel is unloaded a row at a time, with that row being pushed into the grid cage.

Criticality Safety Analysis

The following conservative assumptions were incorporated in the calculational model of the Fuel Rod Pre-Stacking Station.

1. The three trays on the positioner table are stacked vertically with a distance of 8.5" between the bottom of one tray and the bottom of the next. Each tray was assumed to contain the largest diameter fuel rods (0.44")D) at an enrichment of 5.0 wt % U235. The fuel rods were assumed to be tightly packed in a hexagonal array. The trays were assumed to be 9" wide and filled to a height of 6.1" containing 312 fuel rods each. Each tray was flooded with water. The fuel, clad, and water were homogenized over the volume of the box.
2. All structural materials were neglected.
3. An external mist of 0.001 g/cc was assumed.
4. The concrete ceiling (4") and floor (16") have been included in the calculation.

The NITAWL and XSDRNPM codes were used to obtain 16-group cross sections from the 123-group GAM-THERMOS library for input to KENO-IV, the code which was used to determine reactivity of the Fuel Rod Pre-Stacking Station under the conditions noted above. A $k_{eff} = 0.8475 \pm .0054$, was obtained for an infinite array of systems 21.0" center-to-center in the horizontal direction.

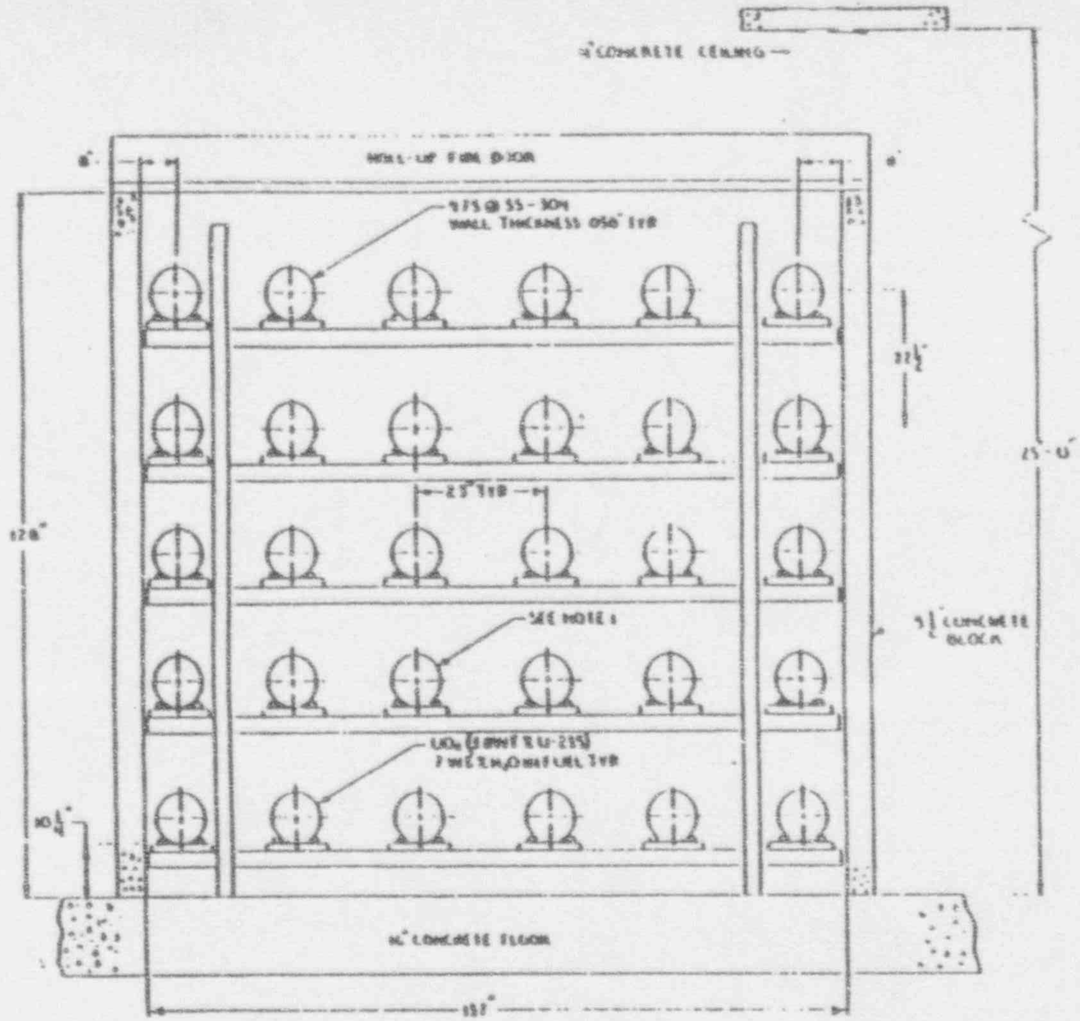
The fuel in the pre-stacked array is in a close packed hexagonal array with 5 internal positions, with 4 fuel rods missing per position. Since the array has much less of a water to fuel ratio than a fuel assembly, even if the 20 fuel position were filled with water within the array, the reactivity would be less than that of a fuel assembly.

The fuel rod box with a pre-stacked array in the rod storage area will be less reactive than the array analyzed in Section 8.5.4 in that the fuel rod box has less fuel per box, 269 fuel rods versus 371 fuel rods. The Fuel Rod Storage Area is dry. Therefore, introduction of water in the void space in the pre-stacked array is not credible.

The individual fuel rod tray which is pulled into the fuel assembly room is again less reactive than a fuel assembly. Only two fuel rod storage boxes can be pulled into the room in fixed positions which are separated by over 4 ft.

8.6 High Enriched Uranium

Up to 350 gms U235 of <20% enriched uranium compounds may be allowed in Building #17 and #21 for purposes of evaluation, analysis, or waste management which consists of scanning drums in preparation for their burial. Such material will be transferred, controlled, and accounted for in accordance with currently approved nuclear material control plans, and except for the drums, all material will be placed in discrete locations specifically designated and posted for this material. None of these materials will be processed through manufacturing operations in Building #17 and #21.



VIRGIN POWDER STORAGE AREA

FIGURE 8.1

VIRGIN POWDER STORAGE AREA

8 7 6 5 4 3 2 1

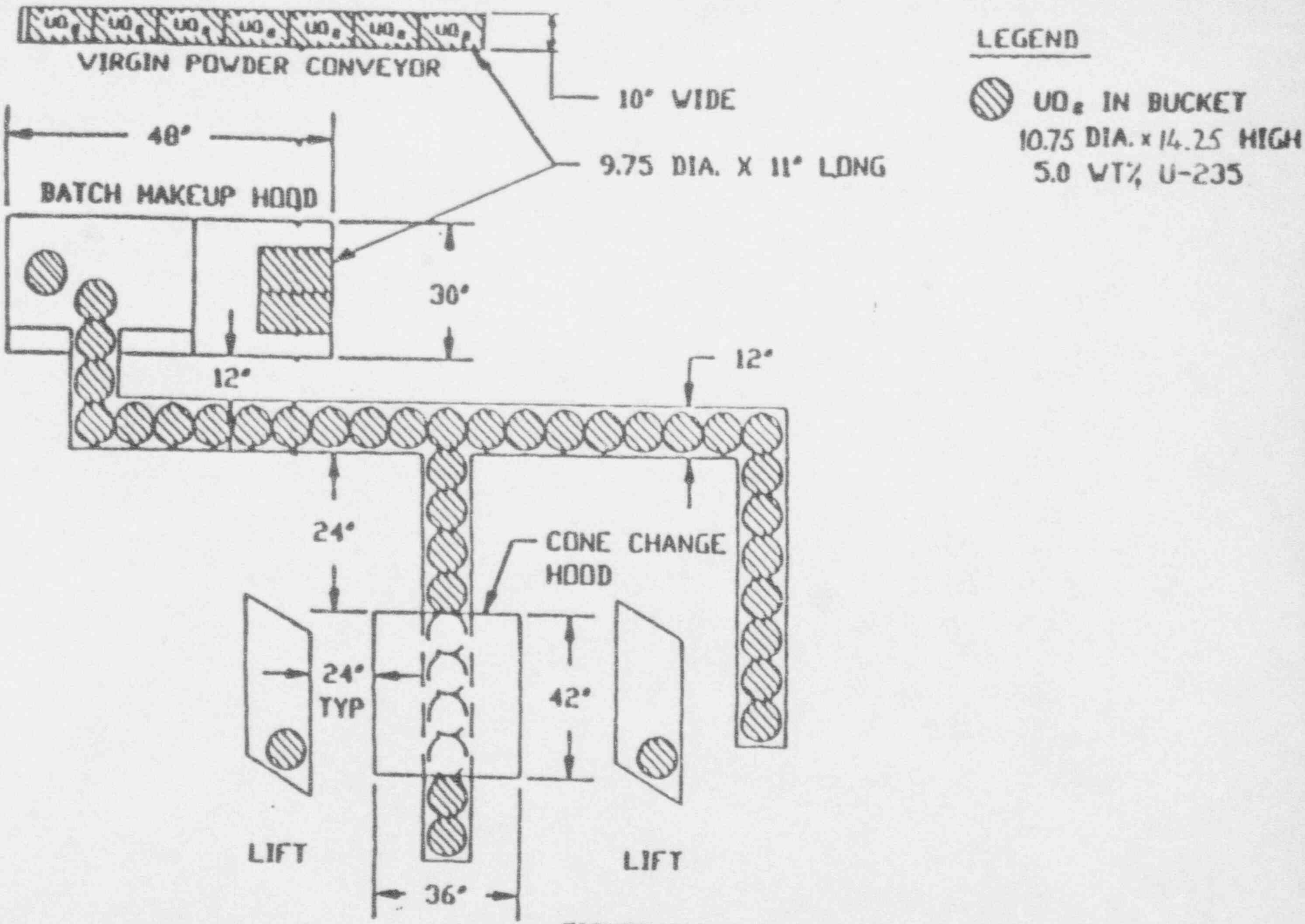


FIGURE 0.2

BATCH MAKEUP HOOD

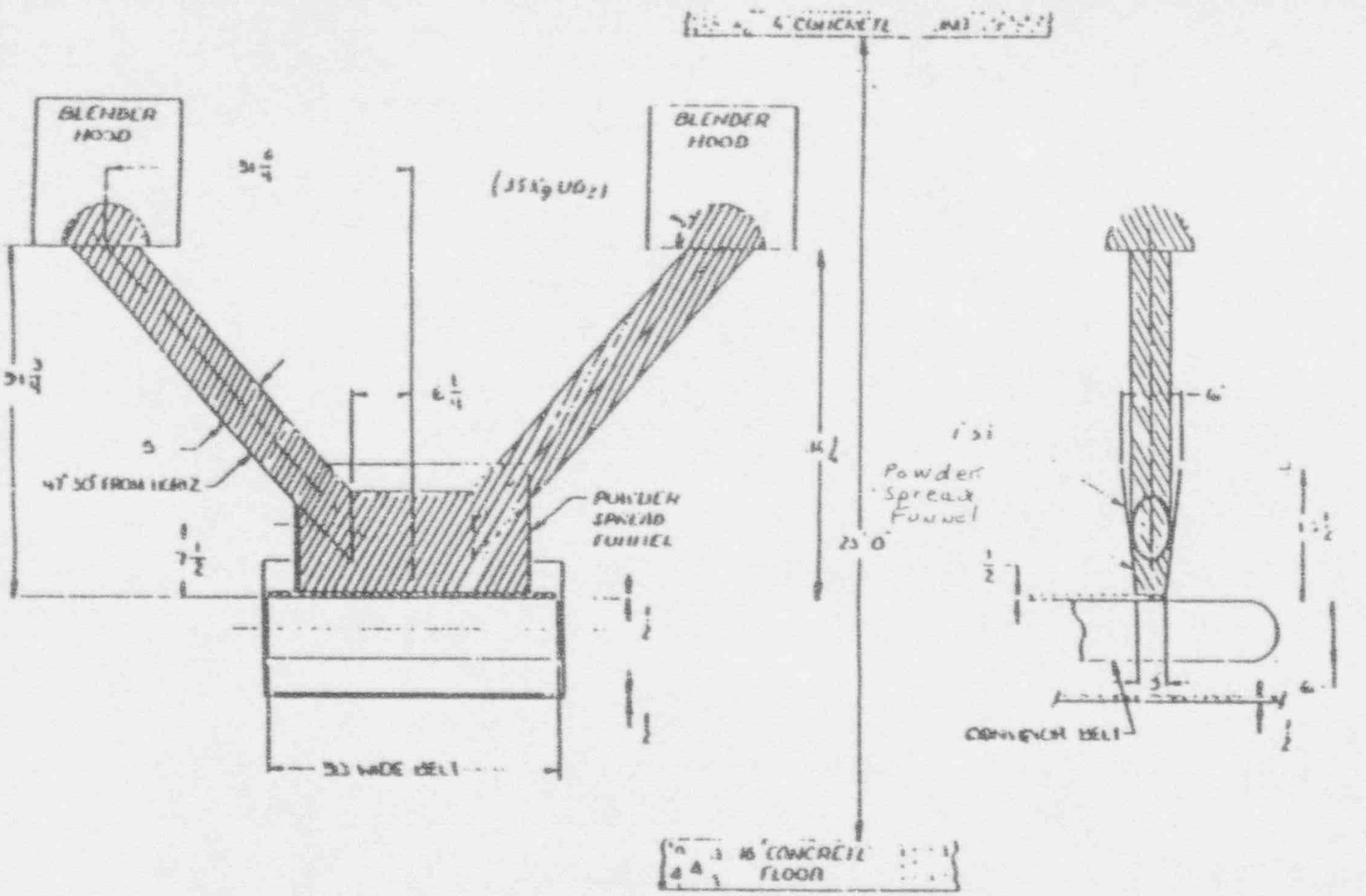


FIGURE 8.3
POWDER PREPARATION STATION-RACK END

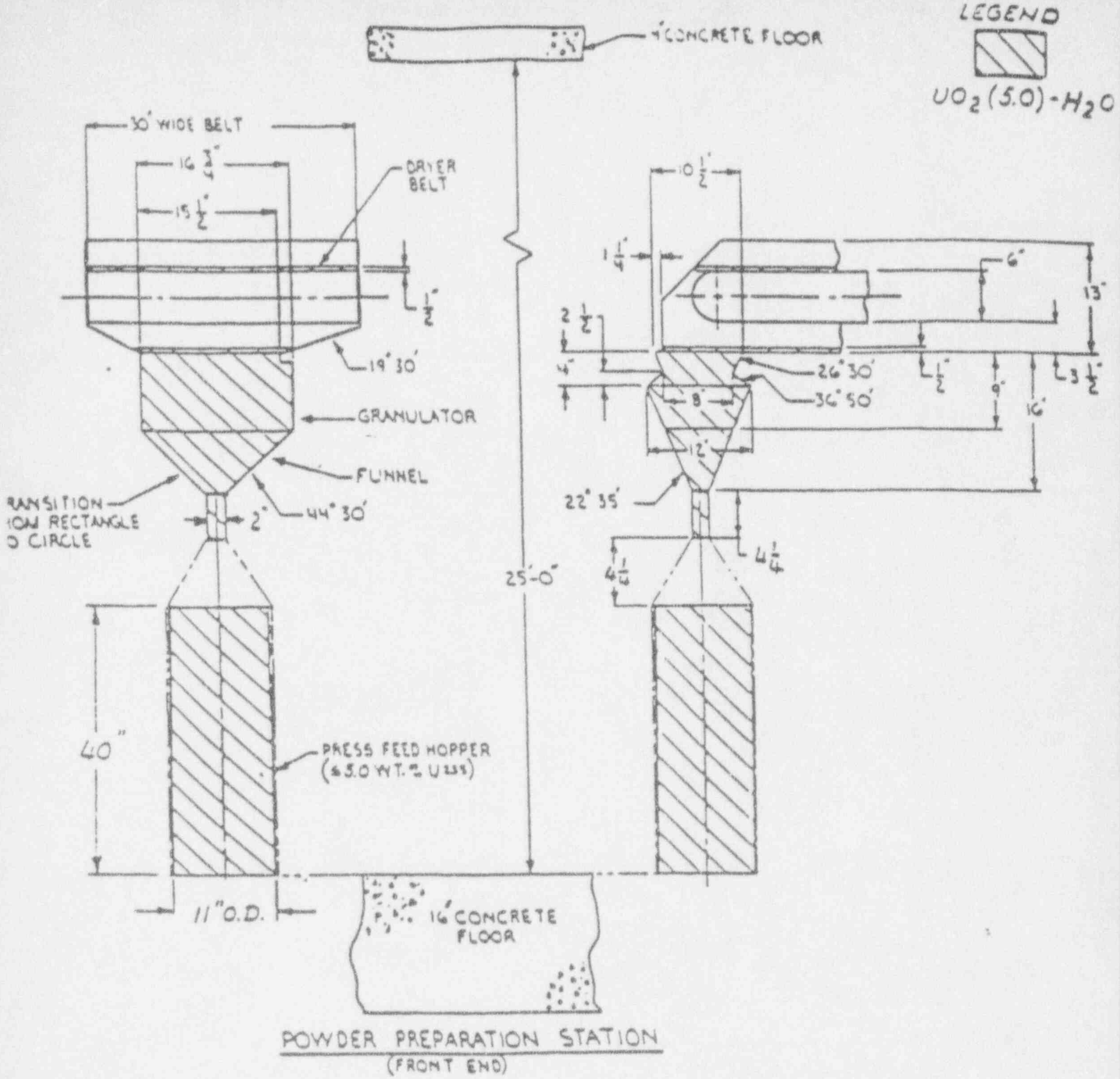


FIGURE 8.4

POWDER PREPARATION STATION - FRONT END

CRITICALITY MODEL

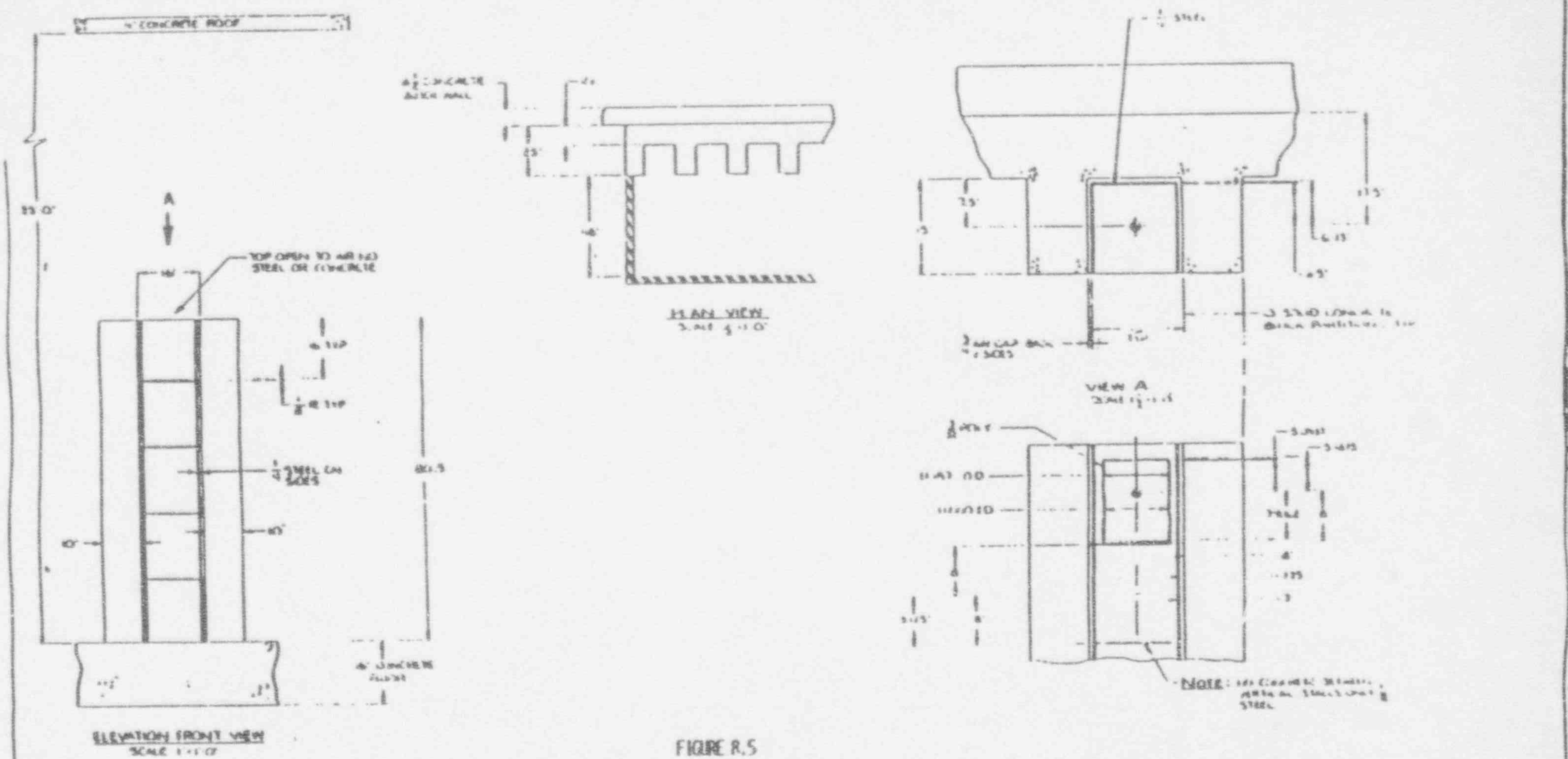


FIGURE R.5

CONCRETE BLOCK STORAGE AREA

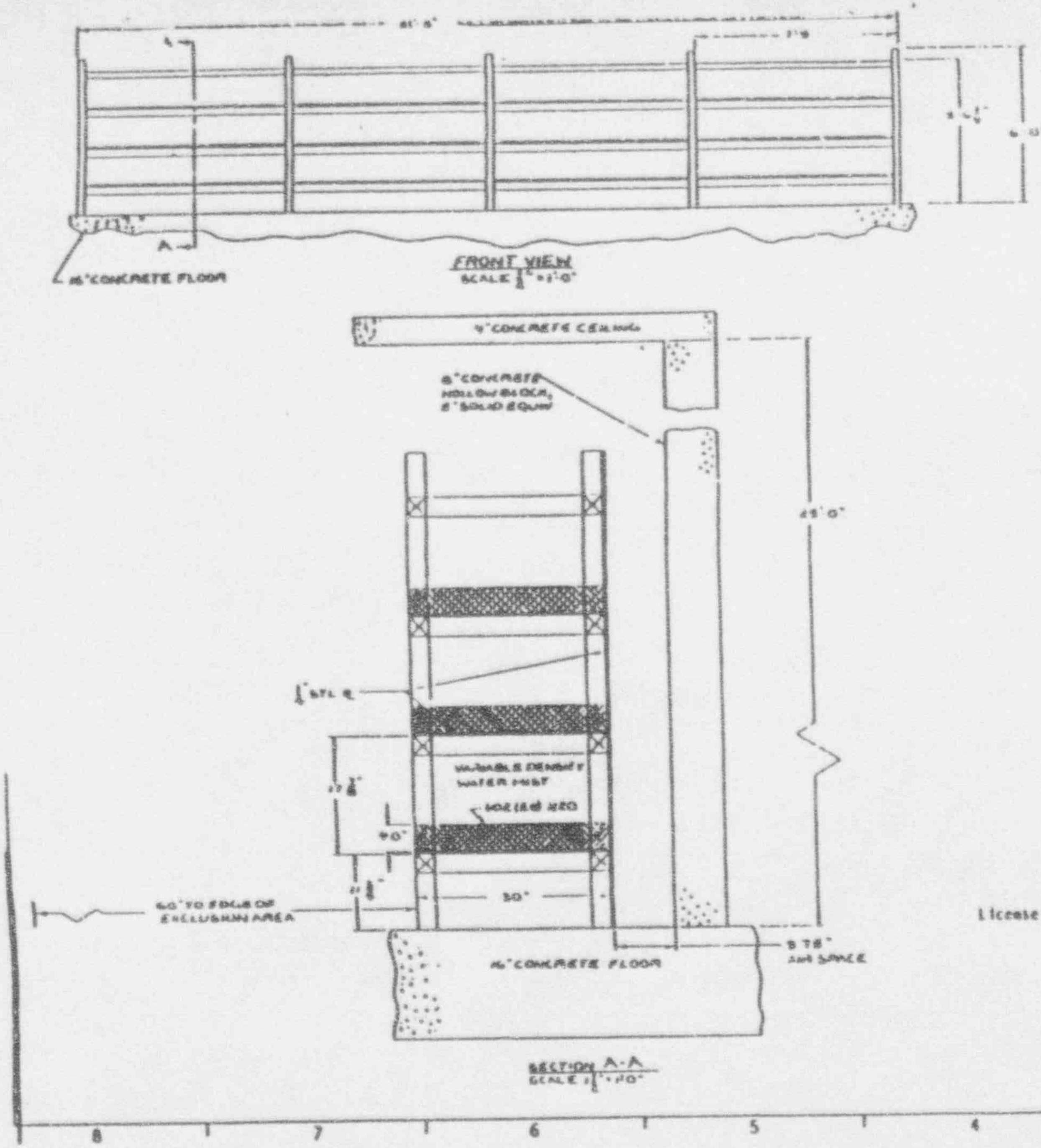


FIGURE 8.6
PELLET STORAGE SHELVES

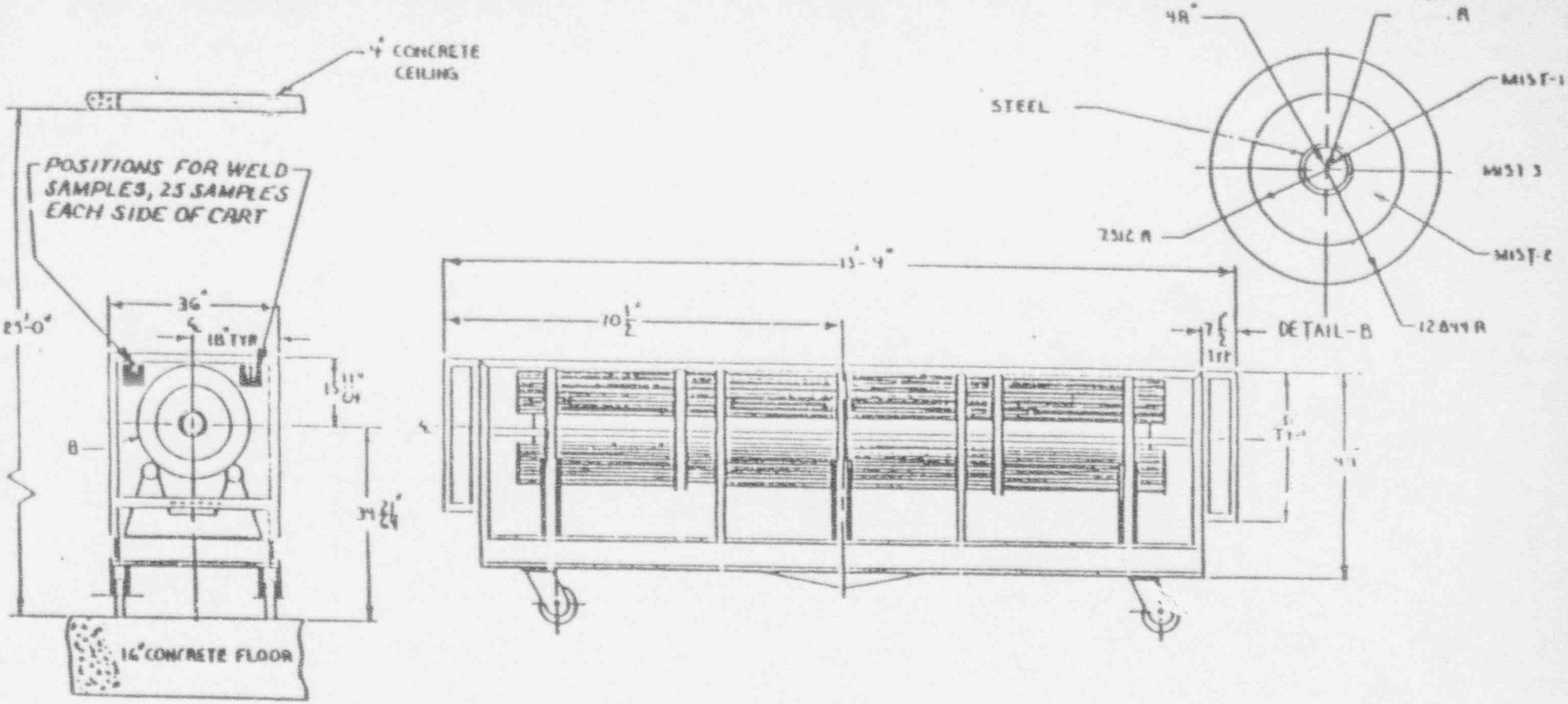


FIGURE 8.7

FUEL TRANS. CART ASSEMBLY

SCALE: $\frac{3}{4}$" = 1'

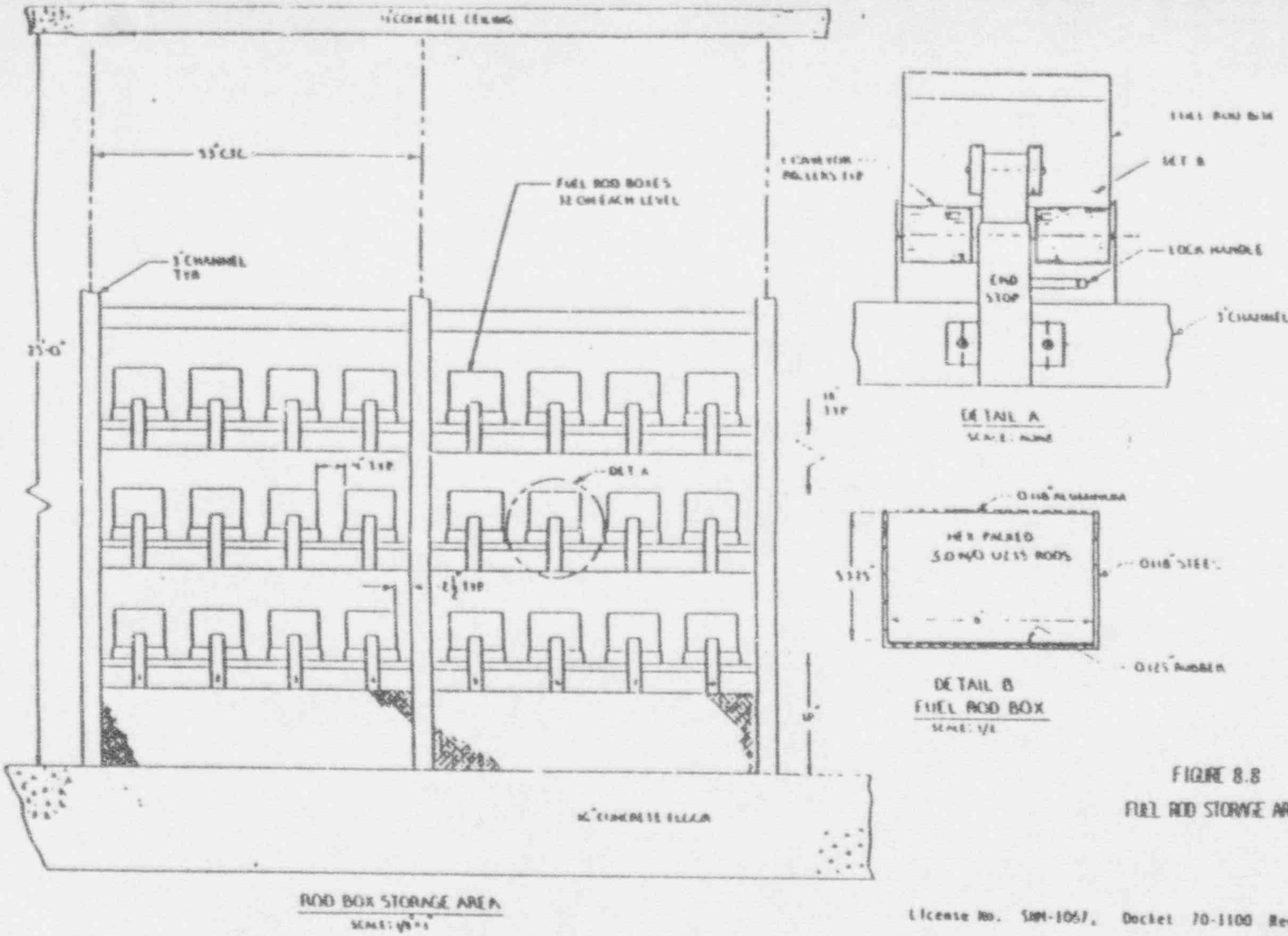
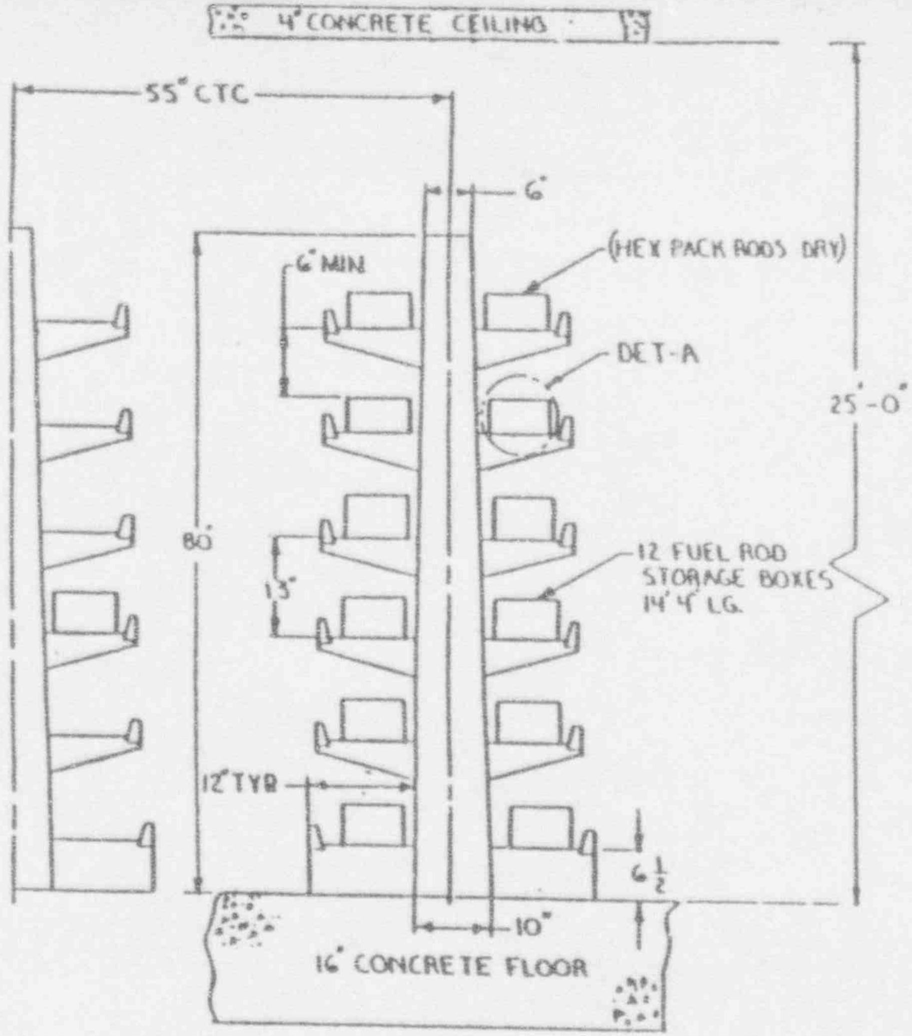
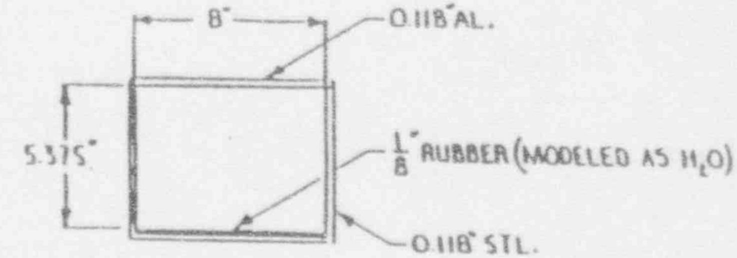


FIGURE 8.8

FUEL ROD STORAGE AREA



ROD STORAGE RACKS
SCALE: 1" = 1'



DET-A
SCALE: 1/4" = 1"

FIGURE 8.9
DOUBLE SHELF ROD STORAGE RACKS

Figure 8.10
Design Parameters of Fuel Assemblies

Fuel Assembly	14 x 14	16 x 16
Fuel Rod Array per Assembly	14 x 14	16 x 16
Total No. Fuel Rod Positions per Assembly	176	236
Fuel Assembly Pitch, in.	8.180	8.116
Fuel Rod Pitch, in	0.580	0.506
Fuel Rod		
Clad Material	Zr-4	Zr-4
Clad O.D., in.	0.440	0.382
Clad Thickness, in.	0.028	0.025
Diametrical Gap, in.	0.0075	0.0070
Active Length, in.	136.7	150.0
Total Length, in.	146.963	161.5
Fuel Pellet		
Material	UO2	UO2
Dish Depth, in.	0.015	0.019
Diameter, in.	0.03765	0.325
Length, in.	0.450	0.390
Density, g/cc/% Theoretical	10.412/95.0	10.41/95.0
Density Stacked, g/cc/% Theoretical	10.03/91.5	10.03/91.5
Spacer Grid		
Material	Zr-4	Zr-4
No. per Assembly	8	11
		Inconel
		1
K_{eff}	0.90	0.90

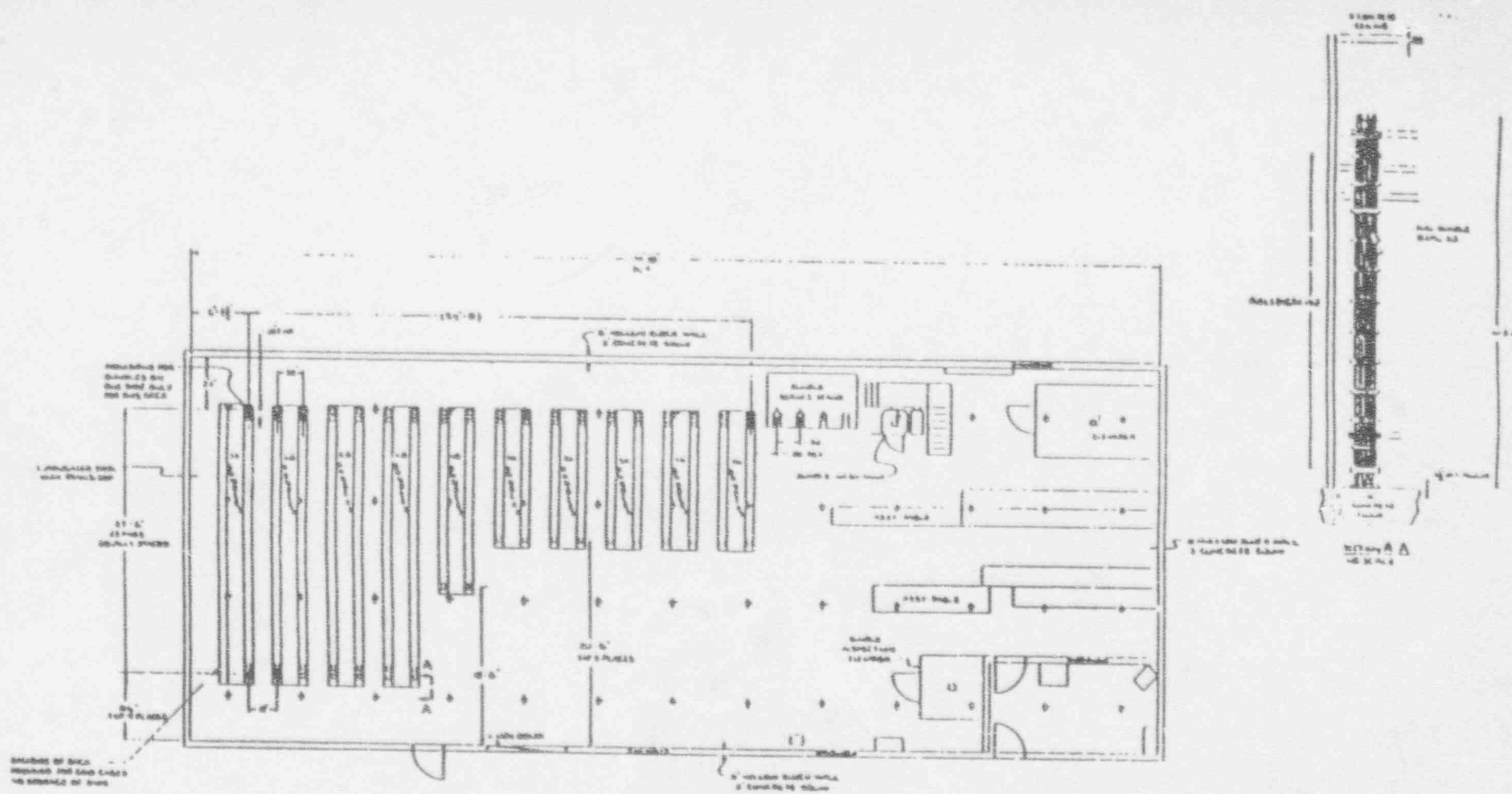


FIGURE 8.11

CITICALITY MODEL

FUEL ASSEMBLY

STORAGE ROOM

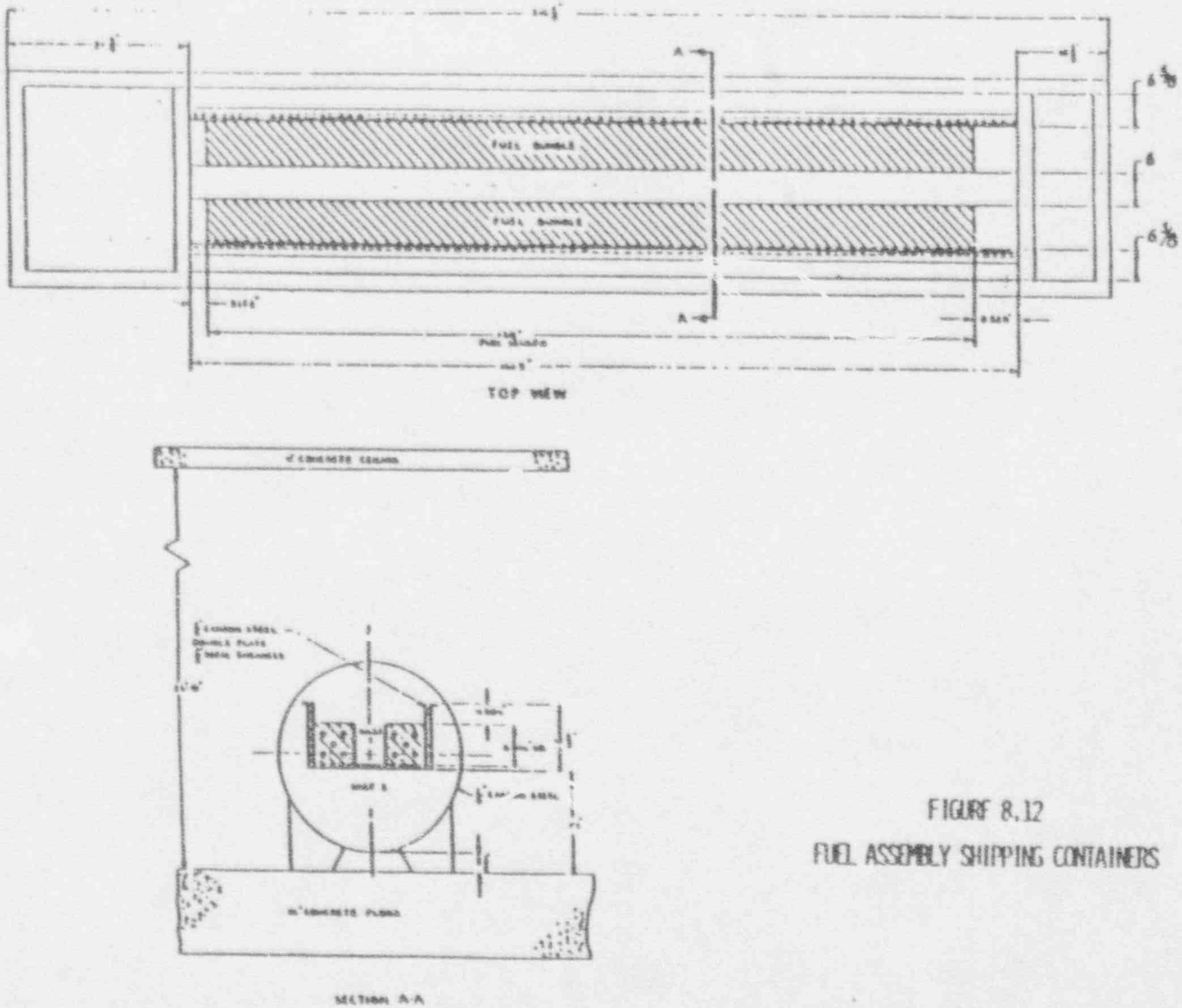
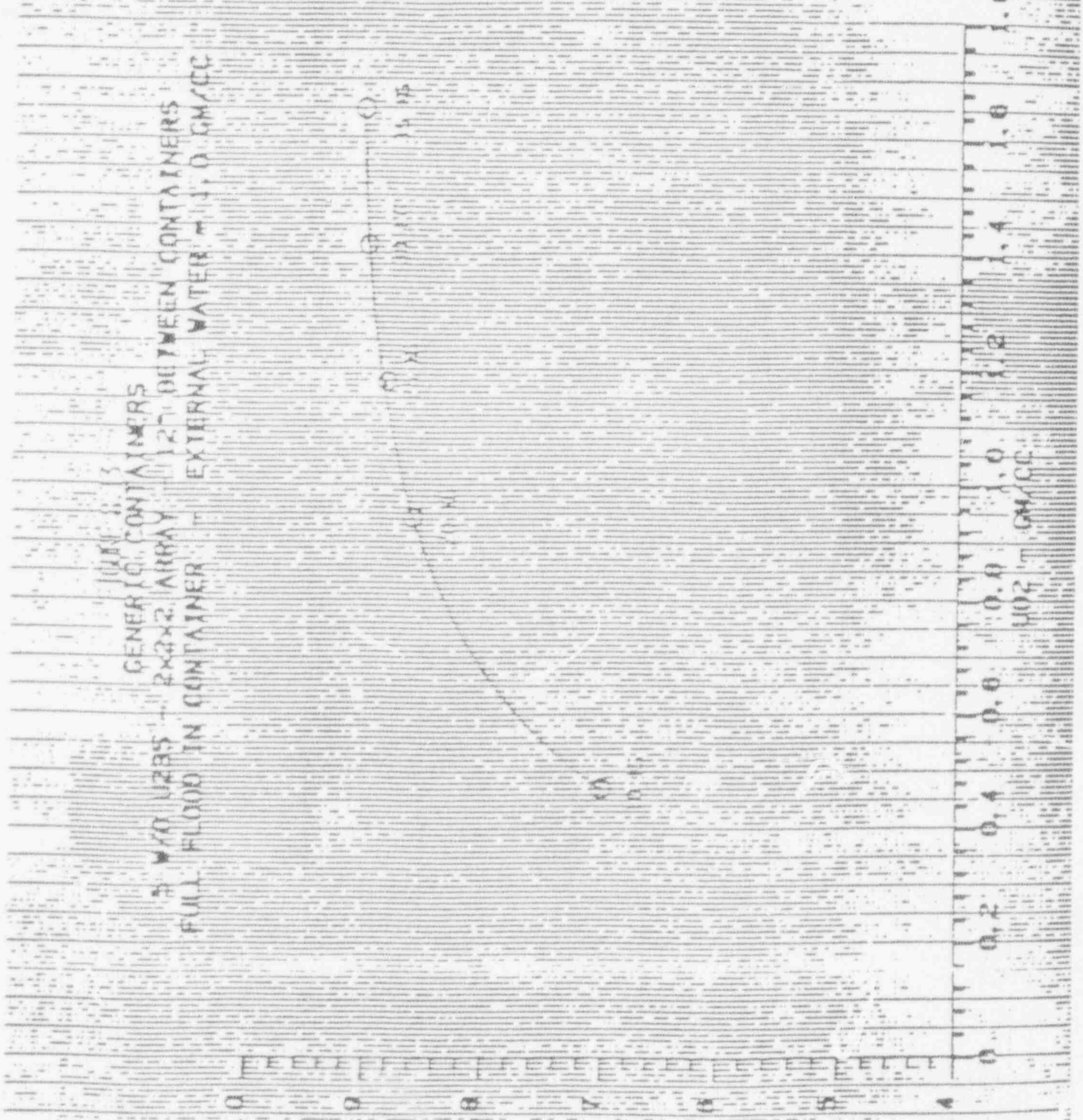


FIGURE 8.12

FUEL ASSEMBLY SHIPPING CONTAINERS



3 W/O U235 - GENERIC CONTAINERS
2x2x2 ARRAY - 12" BETWEEN CONTAINERS
FULL FLOOD IN CONTAINER - EXTERNAL WATER = 1.0 GM/CC
UO2 - GM/CC
KEFF

FIGURE 8.14

CENTRIFUGE COMPLEX LAYOUT IN BUILDING 17
(DIMENSIONS ARE APPROXIMATE)

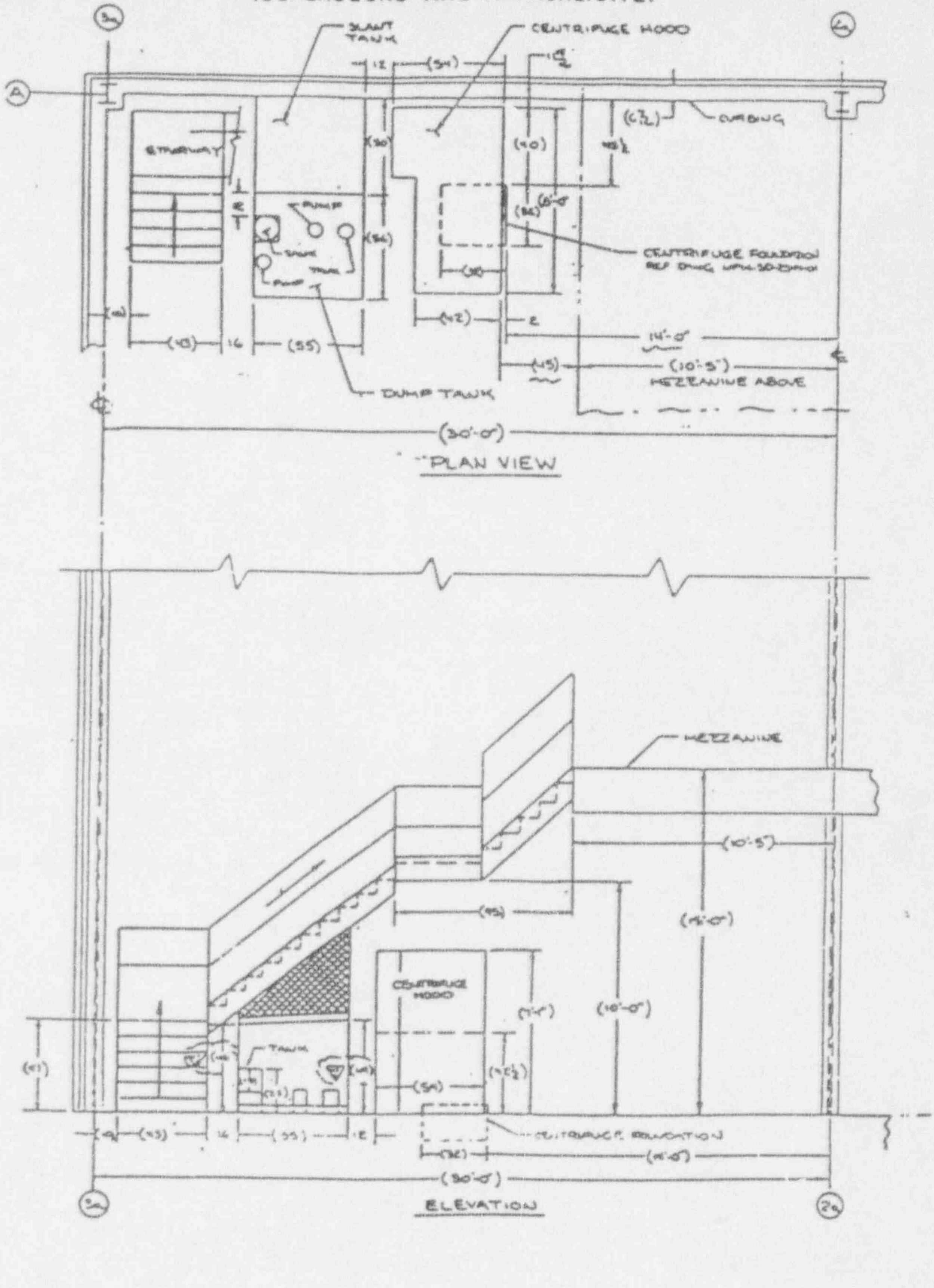


FIGURE 8.15
FLOW CHART FOR THE BUILDING 17 CENTRIFUGE COMPLEX

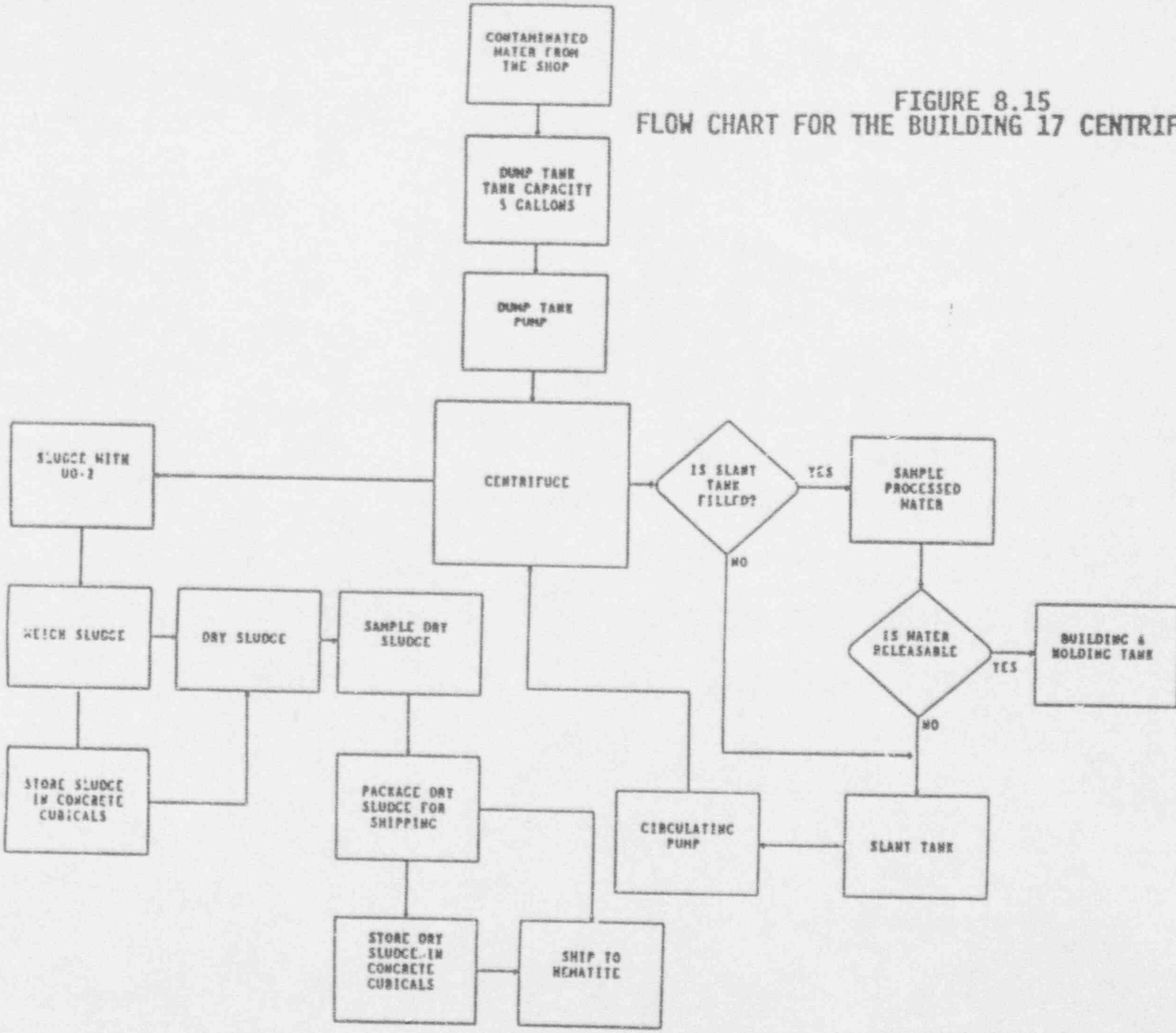
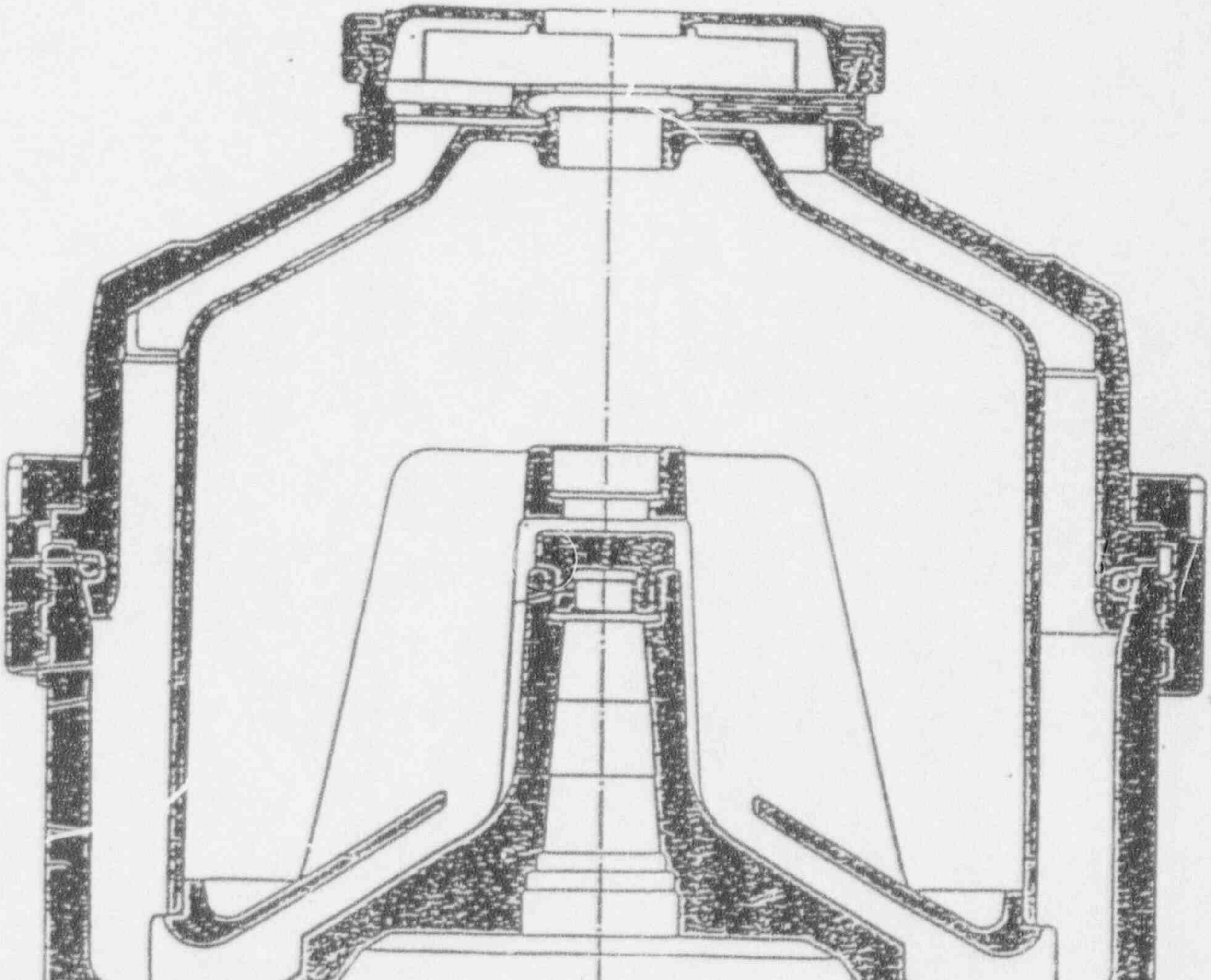


FIGURE 8.16
TWO COMPARTMENT BOWL OF WESTFALIA CLARIFER



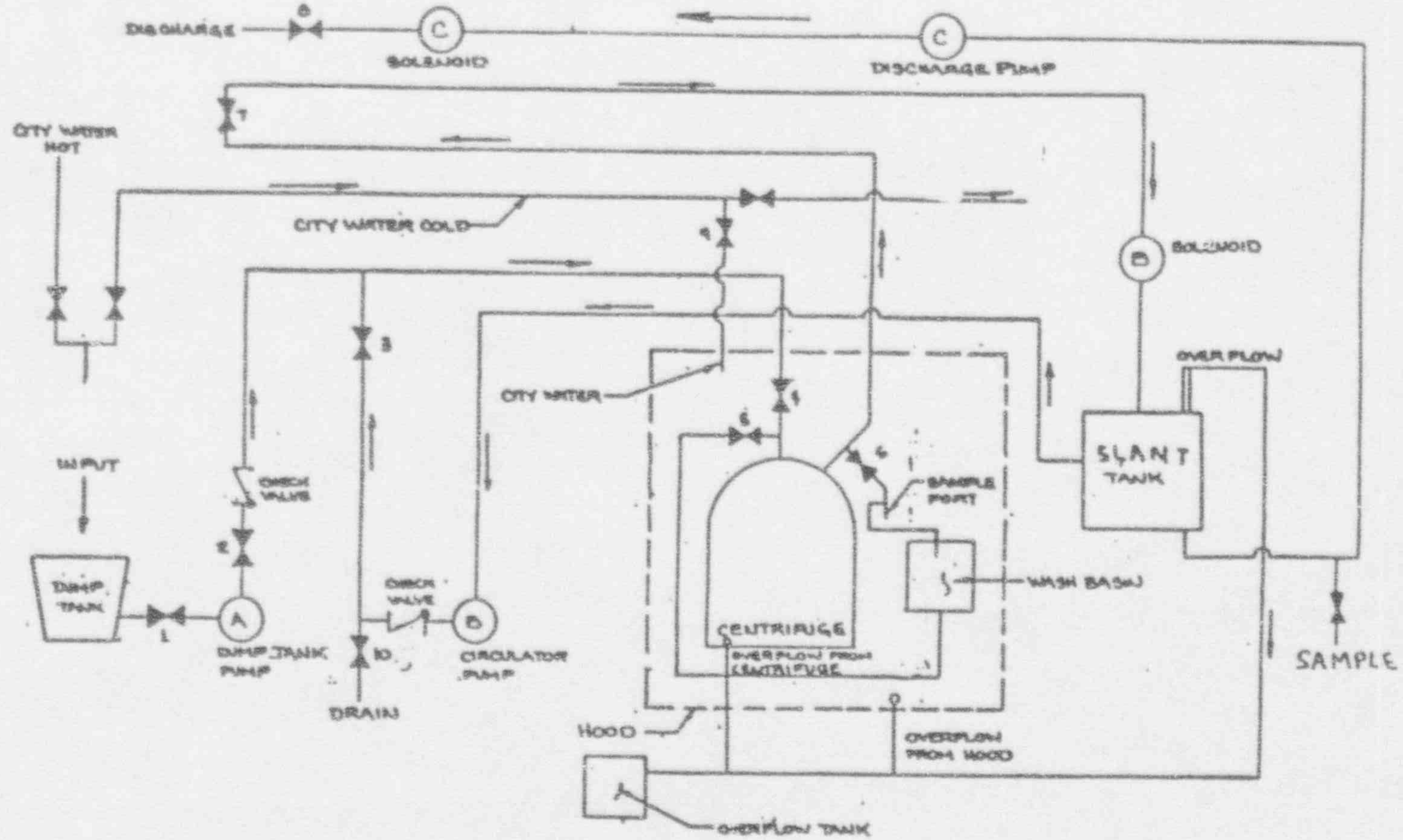


FIGURE 8.17 PIPING LAYOUT DIAGRAM FOR CENTRIFUGE COMPLEX

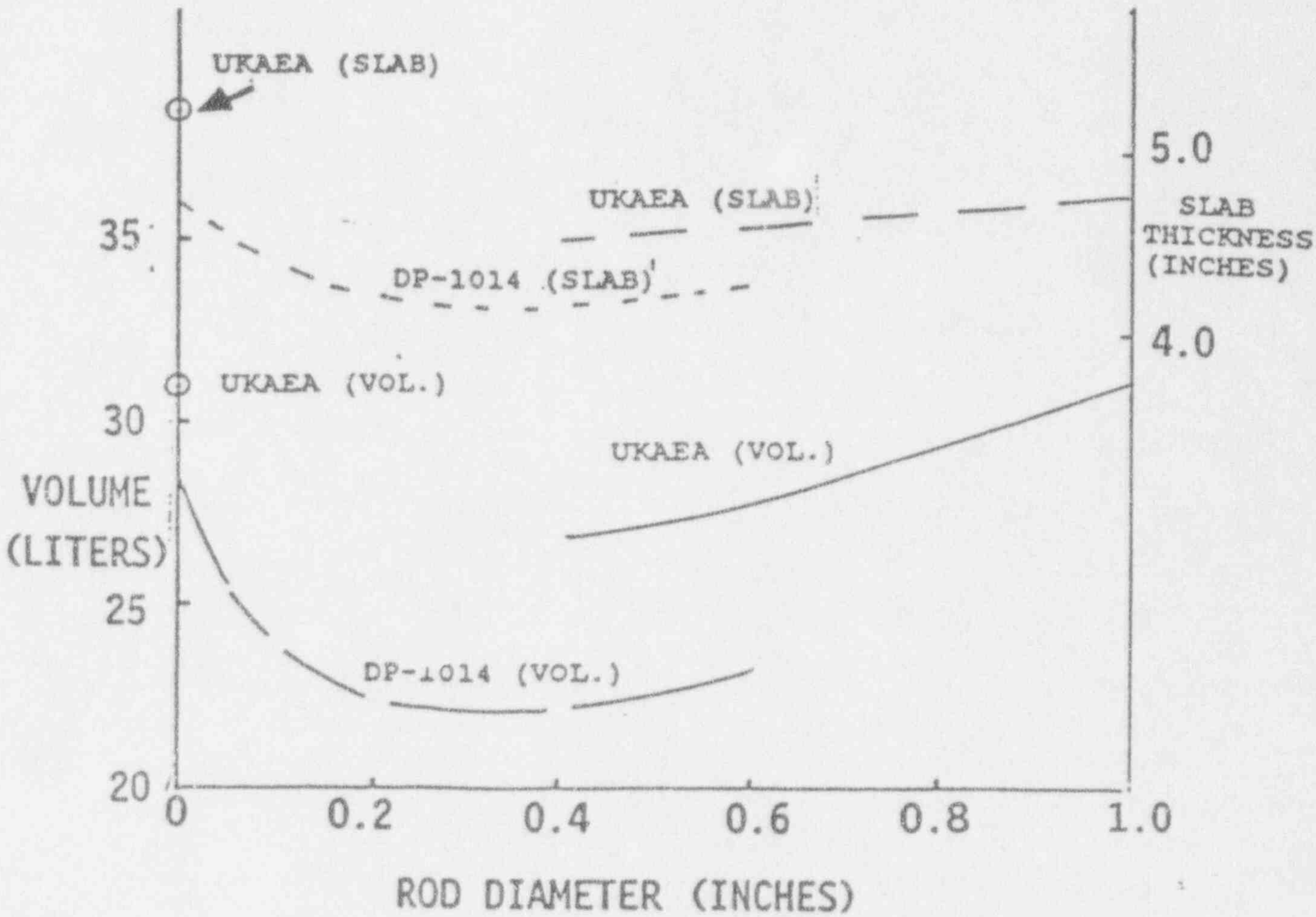

UKAEA (SLAB)
UKAEA (SLAB)
DP-1014 (SLAB)
UKAEA (VOL.)
UKAEA (VOL.)
DP-1014 (VOL.)
VOLUME (LITERS)
SLAB THICKNESS (INCHES)
5.0
4.0
35
30
25
20
0
0.2
0.4
0.6
0.8
1.0
ROD DIAMETER (INCHES)

FIGURE 8.18

SPECIAL PAGE NOTE

(C-E INTERNAL USE ONLY)

Pages II.8.45 through II.8.58 deleted per Amendment 14 to License SNM-1067

The material on this page and all other pages up to and including page II.8-78 has been deleted and replaced with material shown on pages II.8-60 through II.8-74.

Section 8.7

COMBUSTION ENGINEERING, INC.

Windsor, Connecticut

AN ESTIMATION OF THE WATER VOLUME FRACTION PROVIDED IN THE ASSEMBLY ROOM OF BUILDING NO. 17, WINDSOR FACILITY, COMBUSTION ENGINEERING, INC.

FACTORY MUTUAL RESEARCH CORPORATION

S2148.93

ABSTRACT

A method is described for estimating the water volume fraction (water discharge density) provided by automatic sprinklers in the Assembly Room of Building 17 at the Windsor facility of Combustion Engineering, Inc. Water volume fraction in three selected regions in the Assembly Room were evaluated separately. The volume fraction was estimated to be about 0.0075% of the space volume in the three selected regions.

FACTORY MUTUAL RESEARCH CORPORATION

S2148.93

The objective of this project was to estimate the water volume fraction in air which can be provided by the sprinkler system in the Assembly Room of Building 17 at the Windsor facility of Combustion Engineering, Inc.

1. SCOPE OF ESTIMATION

The estimation was performed exclusively for the sprinkler systems and room configuration shown in Figure 1.

The sprinkler system in Figure 1 is in conformance with the NFPA Standard for Sprinkler Systems. The system was installed according to the pipe schedule for ordinary hazard occupancy. The water volume fraction in air was estimated separately for Regions A, B, and C indicated in Figure 1. These regions are delineated by the walls and dashed lines shown in the Figure.

2. ASSUMPTIONS OF ESTIMATION

The estimation was based on the following assumption:

1) Both the diesel and electrical pumps are running to provide sprinkler water.

2) The vertical distance from the base of the riser at Building 17 to the elevation of the sprinklers is about 27 ft, which is equivalent to an elevation head difference of 11.7 psi.

3) The water discharge rate in a region of interest can be obtained from the water supply test data of Building 17 in conjunction with the Factory Mutual Pipe Schedule Sprinkler System Demand Tables (1).

4) The pressure drop due to friction loss from the top of the riser to the region of interest can be estimated from the Factory Mutual Pipe Friction Loss Tables (2).

5) Water drops are homogeneously distributed in the air of the region of interest.

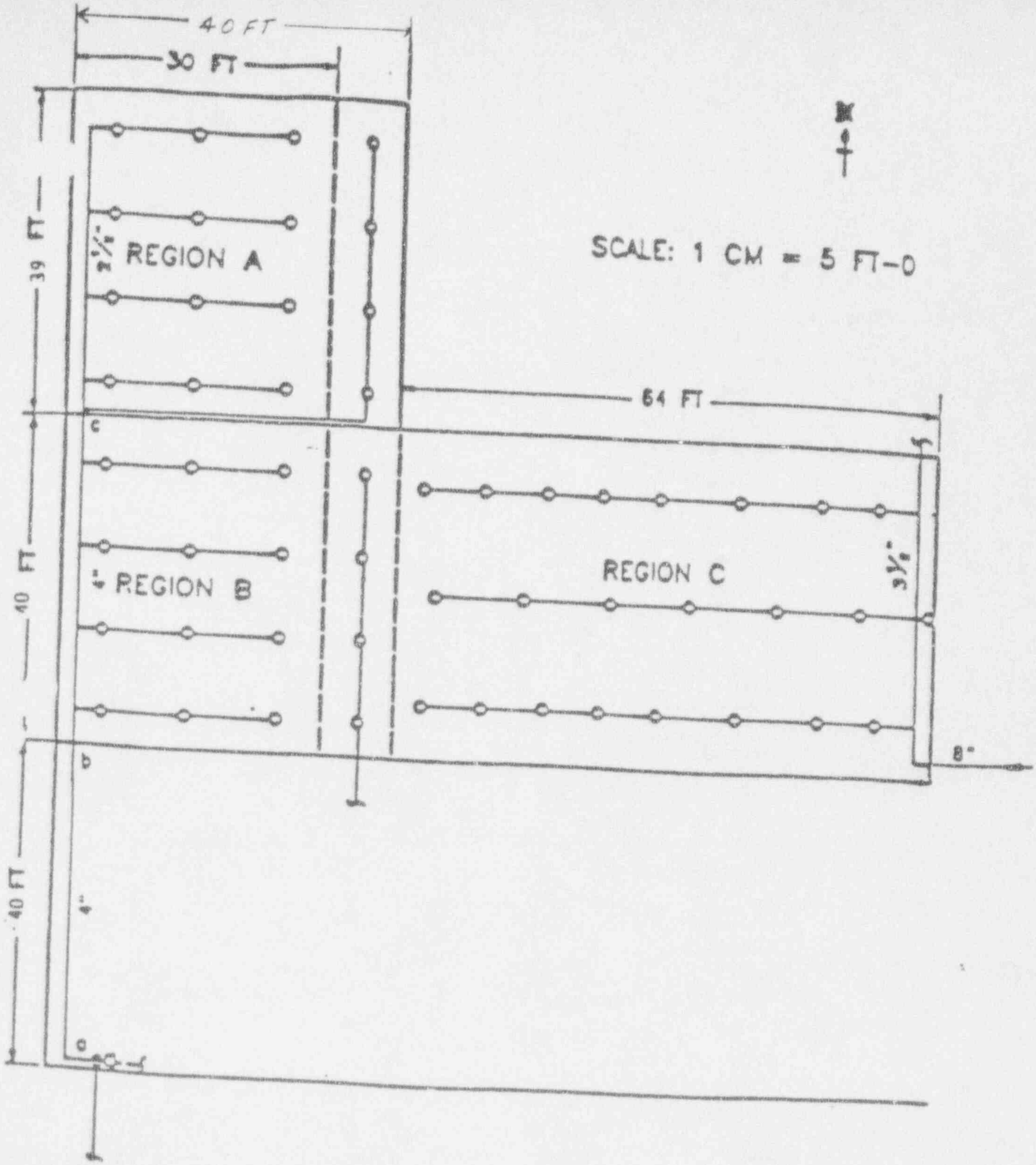


FIGURE 1 SCHEMATIC OF SPRINKLER SYSTEM AND ROOM CONFIGURATION OF THE ASSEMBLY ROOM IN BUILDING NO. 17, WINDSOR FACILITY, COMBUSTION ENGINEERING, INC.

3. PROCEDURE OF ESTIMATION

The procedure used to estimate the water volume fraction in air is described sequentially as follows:

1) Obtain the tabulated water discharge rate from a sprinkler system at the tabulated water pressure at the starting point of the system from the Factory Mutual Pipe Schedule Sprinkler System Demand Tables. In the tables, water discharge rates and water pressures are tabulated such that the water pressure at the end sprinkler in the branch line is 5 psi.

2) Calculate the actual water discharge rate and the corresponding water pressure from the tabulated values obtained in Procedure 1, based on the water supply test data for Building 17 (see Figure 2). For water densities of 0.2 gpm/ft^2 and above, the actual water discharge rate (Q_2) and water pressure (P_2) are related to the tabulated water discharge rate (Q_1) and water pressure (P_1) by

$$(Q_1/Q_2)^{1.85} = P_1/P_2 \ . \quad (1)$$

The water pressure drop due to friction loss from the top of the riser to the region of interest is obtained from the Factory Mutual Pipe Friction Loss Tables.

3) Approximate the actual water pressure at the end sprinkler in the branch line using Eq. (1). Since the water pressure at the end sprinkler is 5 psi in the tables, the actual water pressure is

$$P_2 = 5(Q_2/Q_1)^{1.85} \ . \quad (2)$$

4) Take the average of the water pressure at the starting point of the system of interest and the water pressure at the end sprinkler as the average water pressure of the system.

5) Estimate the volumetric median drop size at the average water pressure of the system.

6) Calculate the water volume fraction in air for the region of interest based on the water discharge rate, space volume in the region, and average vertical downward velocity of the water drops of median size. Use the equation:

$$\text{Water Vol. Frac.} = \frac{\text{(Water Dis. Rate)(Time For Drop to Go From Ceil/Floor)}}{\text{Space Vol. In Region Below Sprinklers}}$$

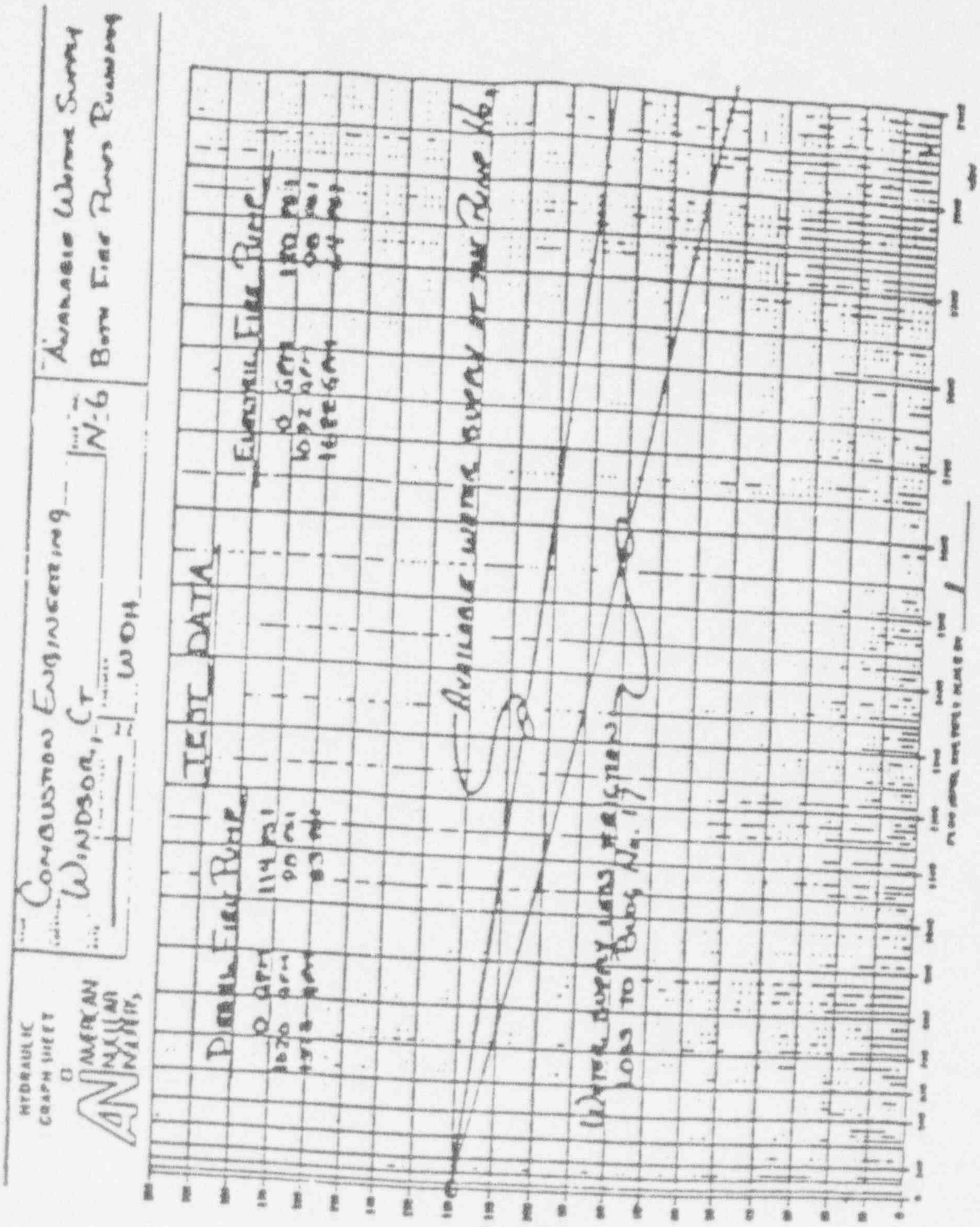


FIGURE 2 WATER SUPPLY TEST DATA FOR BUILDING NO. 17

4. CALCULATIONS

The estimation of water volume fraction in air was performed separately for Regions A, B, and C shown in Figure 1. The following calculation procedures for each region are identified by numbers in accordance with those of Section 3.

Region A

1) From the Factory Mutual Pipe Schedule Sprinkler System Demand Tables for Ordinary Hazard Occupancy, we obtain:

tabulated water discharge rate:	185 gpm
tabulated water pressure:	17
end sprinkler pressure:	5 psi.

2) Actual water discharge rate = 446 gpm
Actual water pressure

$$= (17 \text{ psi}) \times (446 \text{ gpm}/185 \text{ gpm})^{1.85} = 86.6 \text{ psi.}$$

Pressure drop along 80 ft (from a to c in Figure 1) of 3-in. pipe

$$= (0.219 \text{ psi/ft}) \times (80 \text{ ft}) = 17.5 \text{ psi.}$$

Water pressure at the base of the riser

$$= 86.6 \text{ psi} + 17.5 \text{ psi} + 11.7 \text{ psi} = 115.8 \text{ psi.}$$

This pressure agrees with the water supply test data for Building 17 in Figure 2.

3) Actual water pressure at the end sprinkler is

$$P = (5 \text{ psi}) (446 \text{ gpm}/185 \text{ gpm})^{1.85} = 25.5 \text{ psi.}$$

4) Average water pressure in Region A

$$= (86.6 \text{ psi} + 25.5 \text{ psi})/2 = 56.1 \text{ psi.}$$

5) For 1/2-in. sprinklers, the volumetric median drops size at 30 psi is about 0.86 mm (3). Since the median drop size is inversely proportional to the one-third power of water pressure, the median drop size at 56.1 psi is

$$= (0.86\ \text{mm})\ (30\ \text{psi}/56.1\ \text{psi})^{1/3}$$
$$= 0.70\ \text{mm}.$$

6) The downward drop velocity is about 11.5 ft/s for drops of 0.7 mm in diameter (3). The time needed for 0.7-mm drops to fall from the sprinkler to the floor = 27 ft/11.5 ft/s
= 2.35 s.

Therefore, the water volume fraction $= \dfrac{(446)\ (0.039)\ (0.13368)}{(27)\ (30)\ (39)} \times 100$

$= 0.0074\%$

Region B

1) From the tables, obtain:

Tabulated water discharge rate = 185 gpm
Tabulated water pressure = 17 psi
End sprinkler pressure = 5 psi.

2) Actual water discharge rate = 467 gpm

Actual water pressure $= 17\ (467/185)^{1.85}$ psi
$= 94.3$ psi.

Pressure drop along 40 ft (from a to b in Figure 1) of a 3-in. pipe

$= 0.238 \times 40$ psi
$= 9.50$ psi

Water pressure at the base of the riser

$= 94.3 + 9.50 + 11.7$ psi
$= 115.5$ psi.

This agrees with the water supply test data for Building 17.

3) Actual water pressure at the end sprinkler

$= 5\ (467/185)^{1.85}$ psi
$= 27.7$ psi.

4) Average water pressure in Region B

= (94.3 + 27.7)/2 psi
= 61 psi.

5) Median drop size at 61 psi

$= 0.86\ (30/61)^{/13}$ psi
= 0.68 mm.

6) The downward velocity for water drops of 0.68 mm is about 11.5 ft/s.
The time needed for 0.68-mm water drops falling from the sprinkler to the floor = 27/11.5 s
= 2.35 s.

$$\text{The water volume fraction} = \frac{(467)\ (0.039)\ (0.13368)}{(27)\ (30)\ (40)} \times 100$$
$$= 0.0075\%.$$

Region C

1) From the tables, obtain:
Tabulated water discharge rate = 400 gpm
Tabulated water pressure = 17 psi
End sprinkler pressure = 5 psi.

2) Actual water discharge rate = 992 gpm
Actual water pressure $= 17\ (992/400)^{1.85}$ psi
=91.2 psi.

Assume pressure drop sue to friction loss from the base of riser to Region C can be neglected.

Water pressure at the base of the riser

= 91.2 + 11,7 psi
= 102.9 psi

This agrees with the water supply test data for Building 17.

3) Actual water pressure at the end sprinkler

$= 5\ (992/400)^{1.85}$ psi
= 26.8 psi.

4) Average water pressure in Region C

$= (91.2 + 26.8)/2$ psi
$= 59$ psi

5) Median drop size at 59 psi

$= 0.86\ (30/59)^{1/3}$ mm
$= 0.69$ mm.

6) The downward velocity for water drop of 0.69 mm is about 11.5 ft/s. The time needed for 0.69 mm water drops falling from the sprinkler to the floor $= 27/11.5$ s
$= 2.35$ s.

$$\text{The water volume fraction} = \frac{(992)\ (0.039)\ (0.13368)}{(27)\ (64)\ (40)} \times 100$$
$$= 0.0075\%.$$

SUMMARY

The water volume fractions in Regions A, B and C in the Assembly Room of Building 17 (see Figure 1) were estimated separately based on the sprinkler system and room configuration illustrated in Figure 1, and the water supply test data shown in Figure 2. The estimated water volume fractions in air in the above three regions are about 0.0075%.

1) Pipe schedule sprinkler system demand tables, Loss Prevention Data Sheets 2-76, The Factory Mutual System.

2) Pipe friction loss tables, Loss Prevention Data Sheets 2-89, The Factory Mutual System.

3) You, H. Z., "Sprinkler Drop-Size Measurements, Part II: An Investigation of the Spray Patterns of Selected Commercial Sprinklers with the FMRC PMS Droplet Measuring System," FMRC Technical Report, J.I. OG1E7.RA, 1983.

SECTION 8.8

COMBUSTION ENGINEERING, INC.

WINDSOR, CONNECTICUT

AN ESTIMATION OF THE WATER FILM THICKNESS ON FUEL RODS (IN FUEL BUNDLES) DURING A RELEASE OF WATER FROM THE SPRINKLER SYSTEM.

INTRODUCTION

The following are the calculations used to determine water film thickness on fuel rods (in fuel bundles) in storage when the storage room sprinkler system is activated.

The following assumptions have been made:

- No effect due to grids in the fuel bundles
- All water drops falling on the fuel bundle accumulate at the top of the fuel bundles and flow along the fuel rod surfaces.
- Water distribution is uniform in the fuel bundle.

BASIC INFORMATION

Fuel Arrangement (Geometry)

Fuel O.D. = 0.382 inches

Fuel Pitch = 0.506 inches

Flow Rate

For Region B of the storage room = 467 gal/min. (See Section 8.7)

Area of storage room = 30 x 40 = 1200 ft^2

Physical Properties

at 14.7 psia and 77°F

Water Density ρ = 62.3 lb/ft^3

Water Viscosity $\mu = 2.0 \times 10^{-5}\ lb_F\text{-sec}/ft^2$

at 14.7 psia and 50°F

Water Density ρ = 62.3 lb/ft^3

Water Viscosity $\mu = 2.73 \times 10^{-5}\ lb_F\text{-sec}/ft^2$

Area of A Single Fuel Lattice

$A = (0.506)^2/144 = 0.00178\ ft^2$

Clad Perimeter of A Single Fuel Lattice

$\ell = (0.382)/12 = 0.1$ ft

Water Flow Rate Per Fuel Lattice

$$\frac{467\ GPM}{60} \times 0.13368\ \frac{FT^3}{Gal} \times \frac{0.00178\ FT^2}{1200\ FT^2} = 1.54 \times 10^{-6}\ \frac{FT^3}{Sec}$$

Formula for Film Thickness

For this calculation reference 1 and 2 are used.

$$\delta = \left(\frac{3\mu\Gamma g_c}{\rho^2 g}\right)^{1/3}$$

where

δ: film thickness (ft)

μ: viscosity (LB_f-sec/FT^2)

Γ: mass flow rate per unit width of wall (LB/ft sec)

ρ: density (LB/ft^3)

g: acceleration by gravity (ft/sec^2)

g_c: conversion factor

Film Thickness Calculation

at 14.7 psia and 77°F

$$\Gamma = \frac{1.54 \times 10^{-6}\ \frac{FT^3}{Sec} \times 62.3\ \frac{LB}{FT^3}}{0.1\ FT} = 9.6 \times 10^{-4}\ \frac{LB}{FT\text{-}Sec}$$

and

$$\left[\frac{3 \times 2.0 \times 10^{-5}\ \frac{LB_f\text{-}SEC}{FT^2} \times 0.6 \times 10^{-4}\ \frac{LB}{FT\text{-}SEC} \times 32.2\ \frac{FT}{SEC^2}}{(62.3)^2\ \frac{LB}{FT^3} \times 32.2\ \frac{FT}{SEC^2}}\right]^{1/3}$$

$= 0.00025$ FT $= 0.00295$ inches $= 0.0075$ CM at 14.7 psia and 50°F

$$\frac{1.54 \times 10^{-6} \frac{FT^3}{SEC} \times 62.3 \frac{LB}{FT^3}}{0.1 \ FT} = 9.6 \times 10^{-4} \frac{LB}{FT\text{-}SEC}$$

and $= \left[\dfrac{3 \times 2.73 \times 10^{-5} \frac{LB\text{-}SEC}{FT^2} \times 9.6 \times 10^{-4} \frac{LB}{FT\text{-}SEC} \times 32.2 \frac{FT}{SEC^2}}{(62.3)^2 \frac{LB}{FT^2} \times 32.2 \frac{FT}{SEC^2}} \right]^{1/3}$

$= 0.00027$ FT $= 0.0033$ inches $= 0.0083$ CM

Discussion

The above approach is based on laminar film flow and derived theoretically. A more comprehensive approach considering turbulent flow was presented by Dukler (Ref. 3). Dukler shows, in Reference 3, that his approach gives similar film thicknesses as the Nusselt approach (Ref. 1) at zero shear mass at the film/air interface and low Reynolds numbers (less than 300).

For the present case, the Reynolds number is:

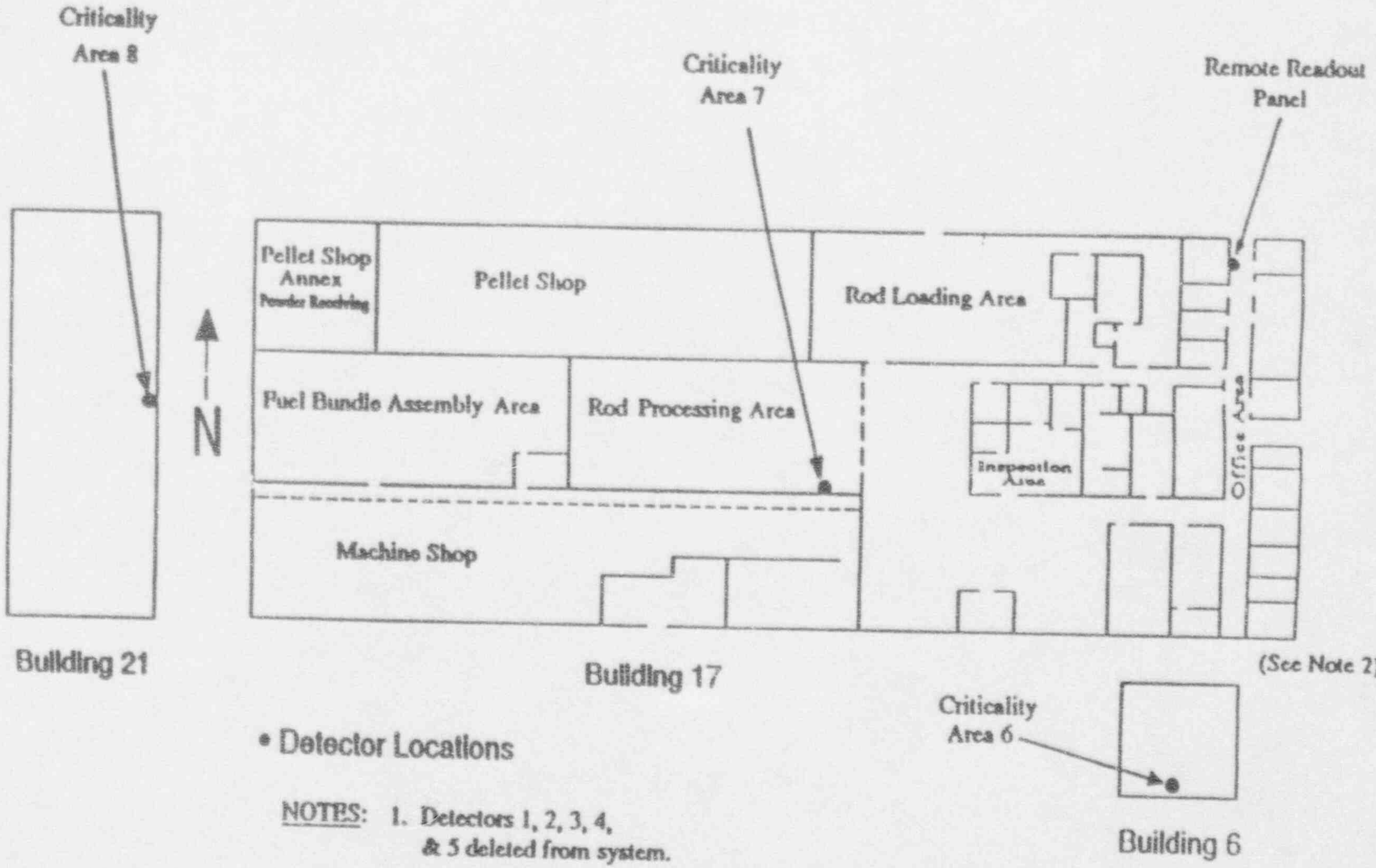
$$Re = \frac{4\Gamma}{\nu}$$

$$Re = \frac{4 \times 9.6 \times 10^{-4} \frac{LB}{FT\text{-}SEC}}{2.0 \times 10^{-5} \frac{LB\text{-}SEC}{FT^2} \times 32.2 \frac{FT}{SEC^2}} = 5.9$$

Therefore, it is concluded that the Nusselt approach is a reasonable one.

References

1. Nusselt, W., ZVDI 60, 541 and 569 (1916)
2. Bird, R.B., etal, TRANSPORT PHENOMENA, Wiley & Sons, Inc., N.Y. (1960)
3. Dukler, A.E., Chemical Engineering Progress SYMP Series No. 30, Vol. 56, Page 1 (1960)



NOTES: 1. Detectors 1, 2, 3, 4, & 5 deleted from system.

2. Bldg. 6 is 1/4 mile from the southeast of Bldg. 17.

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9.0 ACCIDENT ANALYSES

9.1 Spectrum and Impact of Accidents Analyzed

Section 6 of "Environmental Impact Information, Nuclear Fuel Fabrication Facility, Nuclear Laboratories", April 1981 evaluates the consequences of all credible accidents. In all cases examined, the probability of a major accident was found to be extremely low. This low probability is derived from the fact that: 1) all process equipment is designed to incorporate permanently engineered safeguards; 2) strict administrative control of production processes is maintained; 3) the double contingency policy is adhered to in the preparation of safety evaluations; and 4) generous safety factors are included in all facility limits.

Off-site impact of the spectrum of accidents discussed in the Environmental Impact Information is shown below:

Accident	Classification	Off-Site Impact
Injured Employee	Notification of Unusual Event	None
Contaminated Employee	Notification of Unusual Event	None
Process Leak or Spill	Alert	None
Fire	Alert	None
Release of 25 uC of Airborne Radioactive Particulates into CE Site Environs	Site Area Emergency	50% of MPC for insoluble U235 at site boundary
Criticality Accident	Site Area Emergency	Whole Body dose 1.184 RAD Thyroid dose 2.93 RAD
Emergency Alert	General Emergency	None (off-site impact from Emergency Alerts which are reclassified into Plant or Site Emergencies are described above).

9.2 Analysis of Postulated Incidents Having Offsite Impact

9.2.1 Criticality Accident

Since the amount of U235 on site is greater than the minimum mass necessary to achieve criticality, it is necessary to consider the possibility of a criticality incident. While such an accident is theoretically possible, it is highly unlikely because of the administrative and operational controls established by C-E over the receipt, use and storage of enriched uranium. In the history of the low-enriched fuel fabrication industry, there never has been a criticality accident associated with fuel preparation or fabrication. There have been four criticality accidents in high enriched scrap recovery operations, but all these involved wet chemical processing. The operations performed on the Windsor site do not involve wet chemical processing.

Fortunately, criticality accidents have occurred so rarely that no statistical analysis of the probability of such an accident has been attempted. Criticality events that have occurred have had no significant environmental impact. Radiation injuries were restricted to individuals directly involved. Fission products were effectively confined to the processing building in which the event occurred. Prompt evacuation of employees upon a criticality alarm would assure no more than minor radiation doses to all except those in the immediate vicinity of the accident.

In estimating the intensity of a criticality accident, it has been assumed that 1,000,000,000,000,000,000 fissions occur producing approximately 8,000,000 calories of heat. The following type of accident conditions would be necessary to cause a nuclear excursion:

9.2.1.1 Moderated Uranium Oxide - Encapsulated

A sufficient quantity of production assemblies, to sustain a nuclear chain reaction when covered with water, could theoretically, but inadvertently be accumulated.

In the above case, the first spike of the nuclear chain reaction could exceed 1,000,000,000,000,000,000 fissions, ejecting the moderating water as steam in a sufficient quantity to render the system subcritical. The release of fission products is not expected because the enriched uranium is encapsulated in zirconium tubing which is designed to withstand a reactor environment.

9.2.1.2 Moderated Uranium Oxide - Unencapsulated

Unencapsulated enriched uranium such as fuel powder, fuel pellets, or fuel sludges could be accumulated, through inadequate administrative controls, in sufficient quantities for a criticality accident. Similar accidents have occurred in the past with high enriched uranium solutions as previously noted. From 1,000,000,000,000,000 fissions could be expected from a single burst releasing fission products.

The fission products which could be released would combine to yield maximum off-site doses of:

Whole Body Dose	1.184 RAD
Thyroid Dose	2.93 RAD

(Calculated in accordance with NRC Regulatory Guide 3.34)

9.2.2 Major Airborne Particulate Release

A major airborne radioactive particulate release is again highly improbable because of the High Efficiency Particulate Air (HEPA) Filters utilized at C-E NFM-Windsor (99.97% Efficient). All air released to the environs is sampled daily to determine any release. Any release that would result in a concentration of airborne UO2, greater than 50% of MPCa for insoluble U235 in a 24 hour period at the site boundary will be considered significant and the State of Connecticut, Office of Civil Preparedness shall then be notified. A person standing at the site boundary continuously (24 hr/day in a concentration equal to 1.0 MPCa General Population) after 1 year will receive a dose of approximately 0.50 rem. This notification level used by C-E NFM-Windsor is a factor of 1000 less than the NRC notification level required by 10 CFR Part 20. However, because of C-E NFM-Windsor's continued low annual airborne effluent release rates, airborne levels exceeding 50% MPCa at the site boundary would indicate a significant increase above normal operating conditions and notification of state agencies is viewed to be appropriate. A release of this magnitude is insignificant from a radiological safety standpoint, but is used to mitigate the consequenses of a potentially large release.

9.3 Worst Case Scenario

The emergency plan scenario chosen was an accidental criticality excursion. This scenario was chosen because it is the only accident that could occur at C-E NFM-Windsor that has potential for any significant off-site impact.

The emergency plan scenario is based on the following: The postulated accident occurs in the powder processing area of the nuclear fuel manufacturing facility, the area in which low enriched UO2 in powder form is pressed into pellets (UO2 in powder form has the greatest potential of being brought to the conditions required for an accidental criticality to occur). The following conditions would have to occur simultaneously:

1) A violation of criticality mass limits by a factor of 2.3 (A safety factor of 2.3 is incorporated in all mass limits).
2) The accidental introduction of a large quantity of water to the above mass of UO2 powder to bring it to optimum moderation conditions.
3) The assembly of the powder and water mixture into an optimum geometrical configuration. A burst of 1,000,000,000,000,000,000 fissions is then assumed to occur.

This is equivalent to a release of about 32 megawatt seconds, which is a much larger excursion than could be expected in any system in a low enriched fuel fabrication facility. To cause an excursion of this magnitude, a very rapid increase in reactivity would be required, which is not credible in the systems in this facility.

Radiation injuries would be restricted to individuals directly involved and personnel within a 10-20 foot radius of the accident. Prompt evacuation of employees by an automatic criticality alarm system would result in minor radiation doses to all except those in the immediate vicinity of the accident.

9.3.1 Whole Body Cloud Dose

The fission product isotopic release and the average energy used in this postulated accident are shown in Table 9.1. The distance to the nearest resident bordering the C-E Windsor site is 640 meters. The atmospheric conditions assumed for this accident were very conservatively chosen to be Pasquill Type F with a windspeed of 1 meter per second blowing directly toward the home of the nearest resident. If 50% atmospheric condition information was available at the site, the calculated dose would be decreased by at least an order of magnitude.

The dose was calculated assuming a semi-infinite cloud surrounding the individual with a radioisotope concentration equivalent to the center line plume. Since the Pasquiall Type F atmospheric stability condition produces a very small plume, the dose is overestimated by a factor of 8 as a result of the semi-infinite cloud assumption.

The delay in the building was done assuming the volatile fission products were instantly mixed within the pellet shop volume of 291,000 cu. ft. It was assumed that 19,331 cu. ft. of the building volume, which is equivalent to 30 seconds flow through the ventilation system, was discharged with no delay of the fission products. The doses at the site boundary were calculated for this release. The next 19,331 cu. ft. volume was delayed for 30 seconds and the doses at the site site boundary calculated. This same procedure was followed until the entire building volume was released. The doses from each 30 second release were added to determine the total dose from the cloud. This method of calculating has two conservative assumptions - 1) no delay is assumed during each 30 second release, and 2) no further dilution of the fission products in the building is assumed for each release. The only other credit taken was for the wake effect of the building which is only 1.5 based on the minimum external area of the building which is 334 square meters.

OFF-SITE DOSES FROM WORST CASE CRITICALITY ACCIDENT

Whole Body Dose	1.184 RAD
*Thyroid	2.93 RAD

* Calculated using a breathing rate of 0.000347 cu. meters/sec.

The methods utilized in the assessment of this accident are consistent with those used by the U. S. Nuclear Regulatory Commission.

9.3.2 Conclusion

The emergency plan scenario evaluated here indicates that evacuation as a protective measure not be considered. Considering the extreme conservatism described above, the off-site doses will be below all recommended action guidelines. Combustion Engineering's Windsor site has its own effluent and environmental measurement and monitoring program. Environmental samples have been taken on a regular basis over the past 10 years and their results are available for comparision in an emergency situation.

TABLE 9.1

Radioactivity of Important Nuclides Released From the Postulated Criticality Accident

Nuclide	Curies	Average Gamma Energy (Mev)
Kr-83m	3.7E+0	2.0E-3*
Kr-85m	1.6E+1	1.6E-1
Kr-85	1.5E-4	2.2E-3
Kr-87	1.0E+2	7.8E-1
Kr-88	6.5E+1	2.0E 0
Kr-89	4.1E+3	1.6E 0
Xe-131m	3.8E-4	2.0E-2
Xe-133m	5.5E-2	4.1E-2
Xe-133	1.3E 0	4.6E-2
Xe-135m	1.1E+1	4.3E-2
Xe-135	1.6E+1	2.5E-1
Xe-137	3.8E+3	1.6E-1
Xe-138	1.2E+3	1.1E 0
I-129	4.2E-11	
I-131	1.8E-1	3.8E-1
I-132	6.7E-1	2.2E 0
I-133	3.5E 0	6.1E-1
I-134	4.8E+1	2.6E 0
I-135	1.2E+1	1.5E 0

*E-3 is defined as 10^{-3}

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex: 99297

C-E POWER SYSTEMS

March 23, 1982

SNM-1067, Docket 70-1100 (Windsor)
SNM-33, Docket 70-36 (Hematite)

U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. W. T. Crow, Section Leader
Uranium Fuel Fabrication Section
Fuel Processing & Fabrication Branch
Division of Fuel Cycle & Material Safety

Subject: Decommissioning Plans for CE-Windsor and CE-Hematite Nuclear Fuel Fabrication Facilities

References: 1) Letter from H. V. Lichtenberger (C-E) to L. C. Rouse (NRC), dated January 19, 1979

2) Letter from H. E. Eskridge (C-E) to L. C. Rouse (NRC), dated January 19, 1979

Dear Mr. Crow:

The Decommissioning Plan, Reference 1, for the C-E Windsor Plant is included in Appendix A of SNM-1067 and made a part of that document; the Decommissioning Plan, Reference 2, for the C-E Hematite Plant is included in Appendix A of SNM-33 and made a part of that document.

When the Windsor and Hematite operations are phased out and it becomes necessary to decommission these facilities, Combustion Engineering, Inc., will cover all expenses for such decommissioning costs at the time they are incurred through the use of current revenues.

Sincerely,

J. M. West
Vice President, Nuclear Power Systems Division

LICENSE SNM-1067

COMBUSTION ENGINEERING, INC.

NUCLEAR FUEL MANUFACTURING

WINDSOR SITE

DECOMMISSIONING PLAN

JANUARY 15, 1979

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1.0 INTRODUCTION

Combustion Engineering, Inc., is au orized to conduct operations at its Windsor, Connecticut facilities under NRC License SNM-1067. Although there are no federal regulations currently in effect which would require decontamination and decommissioning of these facilities, in May 1978 NRC imposed License Condition No. 24 to SNM-1067 which requires C-E to "submit a plan for the future decontamination of the places of use and sites authorized by this license so they can be released for unrestricted use". In a letter dated December 8, 1978 NRC further defined the details required in this plan including the need for financial arrangements to insure that the funds will be available and committed to cover the costs of decontamination.

Combustion Engineering, Inc., will, after termination of all licensed activities in the facilities covered by NRC License SNM-1067, comply with License Condition 24 as defined in this plan and decontaminate the facilities as required to release them for unrestricted use.

2.0 PURPOSE

The purpose of this plan is to describe the course of action to be taken by Combustion Engineering, Inc., when it becomes necessary to decommission the Windsor Facility licensed under SNM-1067. The object of the decommissioning action is to prepare the site and facilities involved for release for unrestricted use by the company.

3.0 DECONTAMINATION PLAN

Under SNM-1067, all radioactive materials are stored, processed, or retained in buildings 5, 6, and 17 on the Windsor site. Accordingly, the procedures noted below will be applied to these buildings to decontaminate them to the Levels noted in Paragraph 3.1.1.

3.1 Decontamination Guidelines

3.1.1 Decontamination will be to the levels noted below. In addition, every effort will be made to further reduce contamination levels to as low as reasonably achievable:

	AVERAGE (1)(2)	MAXIMUM (1)(2)	TARGET (1)
Removable Alpha	1,000 dpm/100cm2	1,000 dpm/100cm2	200 dpm/100cm2
Fixed Alpha	5,000 dpm/100cm2	15,000 dpm/100cm2	1,000 dpm/100cm2

NOTES: (1) Reference Annex C, dated 11/76, to Combustion Engineering, Inc., Hematite Site Special Nuclear Material License, SNM-33 "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, or Special Nuclear Material".

(2) Measurements of average contamination shall not be averaged over an area of more than one square meter.

(3) The maximum contamination level applies to an area of not more than 100 cm2.

3.1.2 Radioactive surfaces will not be painted until it is demonstrated that contamination does not exceed the above noted levels.

3.1.3 The radioactivity of the interior surfaces of ducts, pipes, etc., will be determined by taking measurements at all access points, provided contamination at these locations is likely to be representative of interior contamination. If such access locations are not likely to be representative, or if interior surfaces are inaccessible, then such interior surfaces shall be assumed to be contaminated in excess of the above noted levels.

3.1.4 All routine health physics and environmental monitoring programs will continue to be conducted in such a manner as to minimize both personnel exposure and release of radioactive material to the site environs. A comprehensive survey will be conducted upon completion of the project. All required records will be maintained for the period specified by applicable regulations.

3.1.5 Disposal of uranium inventory will proceed simultaneously with the plant decommissioning and decontamination. Scrap recovery areas will be the last areas to be dismantled in order to reclaim any uranium generated from clean-up. Security consistent with the requirements of 10 CFR 73 will be maintained until all radioactive materials, licensed under SNM-1067 have been removed from Buildings 5, 6, and 17.

3.2 Decontamination Procedure

3.2.1 Bulk Removal of Radioactive Material

A physical inventory of the facilities will be conducted for all SNM materials covered under License SNM-1067. Inventoried materials will be packaged and disposed of in accordance with regulations in existence at the time that the facility decommissioning takes place.

3.2.2 Disposal of Equipment

Contaminated equipment will be disposed of by burial or sale.

a) Equipment disposed of by burial will not be decontaminated. Instead, the equipment will be cleaned to remove all bulk radioactive materials. The equipment will be vacuum cleaned and then wiped with damp cloths to remove all remaining radioactive particulate matter. The equipment will then be packaged and transported in accordance with applicable DOT and NRC regulations.

b) Contaminated equipment may be sold for use at another fuel cycle facility. In such instances, all exterior surfaces will be cleaned to levels permissible for restricted areas. The equipment will then be packaged and transported in accordance with applicable DOT and NRC regulations.

c) Equipment may be sold for use at non-nuclear facilities. In such instances, all surfaces, interior and exterior, shall be decontaminated so as not to exceed levels given in Paragraph 3.1.1.

3.2.3 Precleaning of Facilities

Upon removal of inventory covered under SNM-1067 and disposal of equipment, the facilities will be precleaned in the following manner:

a) Floors will be vacuum cleaned and wet scrubbed.

b) Walls will be vacuum cleaned and final wiped with damp cloths.

c) All roof trusses, overhead steel, ceilings, overhead ductwork, piping, etc., will be vacuum cleaned and final wiped with damp cloths.

d) All underground drain lines will be pressure flushed repeatedly with clean water.

3.2.4 Presurvey

After completion of the physical inventory, disposal of equipment, and precleaning of the facilities covered under SNM-1067, a radiological survey will be made of contaminated areas to assess the extent of contamination involved and will include, but not be limited to:

a) Floor samples

b) Roof smear samples

c) Smears of all interior wall surfaces

d) Samples of underground piping

e) Smears of roof trusses and supports

f) Smears of overhead piping, ductwork, lighting, etc.

g) Core samples of earth adjacent to contaminated underground effluent lines.

h) Records and drawings of sample locations will be prepared and maintained.

3.2.5 Liquid Effluent System

Underground contaminated piping that cannot be decontaminated to proper levels will be excavated and removed for burial. If sampling indicates contaminated soil, removal and burial of soil will also be accomplished.

3.2.6 Walls

Where the presurvey results indicate residual contamination in the wall surface, removal of the contaminated areas will ensue. Material removal will be repeated as necessary. When the contamination levels of Paragraph 3.1.1 have been reached, a final cleaning (wet wipedown) will be performed.

3.2.7 Floors

Decontamination of floors will follow. Depending on the extent of contamination, surface cleaning, surface removal, or removal of sections of the floor will be dictated. (In the case of the pellet facilities, removal of portions of the floor may be required).

3.2.8 Ventilation System

The contaminated area ventilation systems will remain intact until all of the above steps have been completed. The systems will then be removed and buried.

3.2.9 Demolition

In the event that it is planned to raze a facility, then one of the two courses of action will be taken:

a) The facility may be wholly decontaminated as outlined above, then leveled. In this case, the walls and roof may be disposed of via sanitary landfill.

b) Surface cleaning, rather than surface removal and/or decontamination of walls may be pursued, in which case the walls will be disposed of via burial.

3.2.10 Sequence

The above events will not necessarily proceed one upon completion of the another. Alternatively, two efforts may proceed simultaneously, such as removal of inventoried materials and the removal of equipment. In general, decontamination of the facilities will follow the above outline.

3.3 Final Decontamination Report and Release

3.3.1 Upon completion of the decontamination of the facilities, a comprehensive radiological survey will be taken. If necessary, additional decontamination will be performed.

3.3.2 When it has been established that the facilities are within the limits specified in Paragraph 3.1.1, a survey report will be sent to the Director, Office of Nuclear Materials Safety and Safeguards, with a copy to the Director of Region I. The survey report will:

a) Identify the facilities

b) Show that reasonable effort has been made to reduce residual contamination below the levels specified in Paragraph 3.1.1.

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

d) State the results of the final survey, in units specified in Paragraph 3.1.1.

3.3.3 The facilities covered under SNM-1067 on the Windsor site will be staffed for a maximum of two months after decontamination and while waiting a formal release from the NRC for unrestricted use.

4.0 ESTIMATED COST FOR DECOMMISSIONING

The total estimated cost for decontaminating and decommissioning the present Windsor facilities covered under SNM-1067 is estimated at $170,000 (in 1978 dollars). A complete breakdown of the cost is shown in Table I. Table I also includes an estimate for the total number of cubic feet of contaminated material/equipment to be disposed of at a licensed commercial site.

TABLE 1

Facilities Covered Under SNM-1007	Waste Material To Be Disposed of in cubic ft	Decontamination Cost in $	Transport/ Burial Cost in $	Total Cost in $
Building #5				
a) Facilities	1,000	$10,000	$2,000	$12,000
b) Equipment	2,000	$10,000	$5,000	$15,000
Sub-Total	3,000 cu. ft.			$27,000
Building #6				
a) Facilities	500	$15,500	$1,500	$17,000
b) Equipment	4,000	$19,000	$9,000	$28,000
Sub-Total	4,500 cu. ft.			$45,000
Building 17				
a) Facilities	1,000	$25,000	$3,000	$28,000
Equipment	12,000	$40,000	$30,000	$70,000
Sub-Total	13,000 cu. ft.			$98,000
TOTALS	20,500 cu. ft.			$170,000

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, 1985 — Commission file number 1-117-2

COMBUSTION ENGINEERING, INC.

(Exact Name of Registrant As Specified In Its Charter)

Delaware	13-1587569
(State or Other Jurisdiction of Incorporation or Organization)	(I.R.S. Employer Identification No.)
900 Long Ridge Road, P.O. Box 9308, Stamford, Connecticut	06904
(Address of Principal Executive Offices)	(Zip Code)

Registrant's telephone number, including area code (203) 329-8771

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:

(Title of Class)

(Title of Class)

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 7, 1986
Common Stock — $1 Par Value	33,223,287

The aggregate market value of the voting stock held by non-affiliates of the registrant on March 7, 1986, was approximately $1,125,000,000.

Documents Incorporated By Reference

The following documents are incorporated by reference:

(1) "Financial Section" of the Annual Report to Shareholders for the year ended December 31, 1985, in response to Items 1(a) and 1(d) and Item 3 of Part I, and Items 5 through 8 of Part II; and in partial response to Item 1(b) and 1(c) of Part I, and Items 14(a), (c) and (d) of Part IV.

(2) The Company's Proxy Statement dated March 14, 1986, in connection with its Annual Meeting of Shareholders to be held on April 22, 1986, in response to Items 11 through 13 and in partial response to Item 10 of Part III.

PART I

ITEM 1. DESCRIPTION OF THE BUSINESS

References to the Company contained herein shall be deemed to refer to the Company and its consolidated subsidiaries. References to the Annual Report shall be deemed to refer to the Annual Report to Shareholders for the year ended December 31, 1985.

Item 1(a) General Development of Business

In September 1985, the Company announced plans to sell major portions of its oil and gas equipment and services operations. Accordingly, the results of those operations have been classified as discontinued operations. See Note 4 of the Notes to Financial Statements on Page 43 of the "Financial Section" of the Annual Report. Unless otherwise indicated, all items included in this Form 10-K relate only to continuing operations.

Reference is made to Note 2 of the Notes to Financial Statements on page 42 of the "Financial Section" of the Annual Report regarding acquisitions.

Item 1(b) Financial Information About Industry Segments

Reference is made to Note 14 of the Notes to Financial Statements on pages 47 to 49 of the "Financial Section" of the Annual Report regarding financial reporting by business segments.

Much of the Company's business, especially that relating to steam generating systems, equipment and services for the electric utility industry and design, engineering and construction services, involves long-term contracts of various types, including fixed price and cost plus fee type contracts with some contracts including variations of both types. Certain contracts include incentive provisions whereby the profit is adjusted depending upon performance. 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 recorded principally on the basis of the estimated stage of completion. However, 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. These contracts extend over a period of from several months to four or more years. Revisions in cost estimates during the progress of the work under long-term contracts have the effect of including in subsequent accounting periods adjustments necessary to reflect the results indicated by the revised estimates of final cost. Projected or realized losses under long-term contracts, if any, are provided for in the period when first determined. See Note 1(d) of the Notes to Financial Statements on page 38 of the "Financial Section" of the Annual Report.

Cost estimates for long-term contracts take into account all anticipated costs including, among others, engineering, manufacturing, subcontracting and field construction costs which are required to meet the specifications, including warranties, of the contracts. In addition, when a long-term contract for steam generating equipment is completed for accounting purposes (usually after payment by the customer of amounts retained under terms of the contract and satisfactory operating performance of the equipment), provision is made for future warranty costs, generally on the basis of past experience.

Item 1(c) Narrative Description of the Business

Reference is made to "Business Segments and Brief Description of the Business" shown on page 51 of the "Financial Section" of the Annual Report regarding a narrative description of the Company's business.

Raw Materials

The principal 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 the business. 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

Reference is made to the "Management Discussion and Analysis of Financial Condition and Results of Operations — Unfilled Orders" on page 33 of the "Financial Section" of the Annual Report. Approximately 48% of the consolidated December 31, 1985 backlog of unfilled orders is expected to be recorded as sales (principally on the percentage of completion method) in 1986 and the remainder in subsequent years. Not included in backlog at December 31, 1985 is the $230 million Detroit resource recovery contract announced in December 1985, pending release of escrowed financing.

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 equipment, products and services for industrial markets, 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.

With respect to steam generating systems, equipment and services for the electric utility industry, the Company is one of the largest domestic manufacturers of fossil fueled steam generating systems and equipment and is one of four domestic manufacturers of nuclear steam supply systems. The competitors for fossil fueled steam generating systems include The Babcock & Wilcox Company, a wholly-owned subsidiary of McDermott International, Inc. and Foster Wheeler Corporation. The other domestic manufacturers of nuclear steam supply systems are Westinghouse Electric Corporation, General Electric Company and The Babcock & Wilcox Company. In addition, there is a substantial amount of competition in foreign markets including Framatone (France), Westinghouse Electric Company and K.W.U. (Germany).

Lummus Crest Inc., the principal component of the design, engineering and construction services segment of the Company, continues to be one of the largest domestic firms engaged in designing, engineering and constructing chemical process plants, petroleum refineries and other industrial facilities. However, the changing nature of the industry has created differing degrees of competition due to a shift in the size and nature of recent contract awards. As a result, the Company now competes with various engineering firms of all sizes, including the major firms which had been our primary competitors in the past.

Usually, the Company competes for new orders by responding to specific invitations to bid. The principal basis 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.

Research and Development

The estimated amount spent during 1985, 1984 and 1983 on Company sponsored research and development activities was $60,062,000, $98,891,000 and $48,284,000, respectively, and on that which was customer sponsored was $19,141,000, $19,649,000 and $23,025,000 respectively.

Compliance with Environmental Protection Laws

Compliance by the Company with Federal, state and local environmental protection laws required capital expenditures of $226,000 in 1985, $983,000 in 1984 and $2,249,000 in 1983. It is estimated that capital expenditures in 1986 for such purposes will be approximately $300,000.

Employees

At December 31, 1985, the Company employed 24,751 persons.

Item 1(d) Financial Information about Foreign and Domestic Operations and Export Sales

Reference is made to Note 14 of the Notes to Financial Statements shown on pages 47 to 49 of the "Financial Section" of the Annual Report.

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 (4)*
Brantford, Ontario (1)
Chattanooga, Tennessee (1) (2) (3)
Chicago, Illinois (1)*
Dry Branch, Georgia (2)#
East Chicago, Indiana (1) (3)
Gabbs, Nevada (2)*#
Houston, Texas (4)*
Maple Grove, Ohio (2)#
Marion, North Carolina (1) (3)
Monongahela, Pennsylvania (1) (3)
Northampton, England (2)*
Ottawa, Ontario (1) (3)*
Paris, France (4)*
Rochester, New York (1)
Sandersville, Georgia (2)#
Sherbrooke, Quebec (1) (3)
Springfield, Ohio (1)
Stamford, Connecticut (Corporate Office)*
Stevenage, England (1)
St. Catherines, Ontario (1) (2)
The Hague, Netherlands (4)*
Valley Forge/King of Prussia, Pennsylvania (1) (2)
Walnut Creek, California (2)
Waterford, Pennsylvania (2)*
Wellsville, New York (1) (3)
Windsor, Connecticut (1) (3)*
Worcester, Massachusetts (1)

(1) Equipment for industrial markets
(2) Products and services for industrial markets
(3) Steam generating systems, equipment and services for the electric utility industry
(4) Design, engineering and construction services

* 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 Financial Statements shown on page 46 of the "Financial Section" of the Annual Report for the year ended December 31, 1985.

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

None

PART II

ITEMS 5. THROUGH 8.

The "Financial Section" of the Annual Report to Shareholders for the year ended December 31, 1985 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	56 to 59
Item 6 — Selected Financial Data	28
Item 7 — Management's Discussion and Analysis of Financial Condition and Results of Operations	29 to 33
Item 8 — Financial Statements and Supplementary Data	34 to 55

ITEM 9. DISAGREEMENTS ON ACCOUNTING AND FINANCIAL DISCLOSURE

None

ITEMS 10 THROUGH 13.

The Company's Proxy Statement dated March 14, 1986 in connection with its Annual Meeting of Shareholders to be held on April 22, 1986, has been filed with the Securities and Exchange Commission and the information set forth under "Executive Compensation" on pages 6 to 12 thereof and information with respect to stock ownership set forth on pages 2 to 5 thereof is hereby incorporated by reference.

Listed below are the officers of the Company.

Name	Age	Position Presently Held
Charles E. Hugel	57	President and Chief Executive Officer
George S. Kimmel	51	Executive Vice President and Chief Financial Officer
Charles E. Barnett	46	Vice President, General Counsel and Secretary
Joseph F. Condon	60	Vice President-International
William J. Connolly	56	Vice President-Corporate and Investor Relations
James B. Kelly	58	Vice President in charge of Industrial Group
Mitchell Kiamie	64	Vice President-Operational Performance and Analysis
Sven A. Kreipke	60	Vice President
Donald E. Lyons	56	Vice President in charge of Power Systems Group
John F. Mangold	59	Vice President in charge of Oil and Gas Group
Robert H. Masson	50	Vice President and Treasurer
Dudley C. Mecum	51	Vice President in charge of Urban Systems and Services, and Engineering and Construction Group
John R. Peterson	52	Vice President-Strategy and Business Development
Jeffrey S. Rubin	42	Vice President and Controller
Dale E. Smith	42	Vice President-Human Resources
Eugene T. Yon	49	Vice President in charge of Engineered Systems and Controls Group

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 Chief Executive Officer in April 1984 and President and Director of the Company effective September 1, 1982. Prior to joining the Company, he was an Executive Vice President of American Telephone & Telegraph and, prior to that, he was the President of Ohio Bell Telephone.

Mr. Kimmel was elected an Executive Vice President in December 1984, Vice President-Finance in June 1980 and a Vice President of the Company in April 1979. He was elected a Director in April 1981.

Mr. Barnett was elected a Vice President 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. Condon was elected a Vice President of the Company in January 1978 and is responsible for coordinating the Company's international manufacturing and licensing activities, sales and market services.

Mr. Connolly was elected a Vice President of the Company in April 1976 and was responsible for corporate marketing and communications until September 1980 when he became Vice President-Corporate and Investor Relations.

Mr. Kelly was elected a Vice President of the Company in April 1967. During the past five years he has been the senior operating officer of the Industrial Group.

Mr. Klamie was elected a Vice President of the Company in August 1967. Prior to April 1984 he was the principal accounting officer of the Company.

Mr. Kreipke was elected a Vice President of the Company in June 1961. Prior to November 1985 he was the senior operating officer of the Engineering and Construction Group.

Mr. Lyons was elected a Vice President of the Company in September 1982. Effective with his election, he was placed in charge of the Power Systems Group. Prior to assuming his present position, he was Vice President of Operations for the Power Systems Group and, prior to that, Vice President of Fossil Power Systems.

Mr. Mangold was elected a Vice President of the Company in January 1982. Effective March 1, 1984 he was placed in charge of the Oil and Gas Group. Prior to that, he had been a senior operating officer of the Process Equipment Group.

Mr. Masson was elected a Vice President of the Company in November 1980. Prior to joining the Company, he was Vice President and Treasurer of PepsiCo, Inc.

Mr. Mecum was elected a Vice President and Director in April 1985. Effective with his election he was placed in charge of the Urban Systems and Services business of the Company. In November 1985 he was also placed in charge of the Engineering and Construction Group. Prior to joining the Company he was managing partner of the New York office of Peat, Marwick, Mitchell and Co. and a member of the firm's operating committee.

Mr. Peterson was elected a Vice President of the Company in September 1980 and is responsible for directing corporate-level programs in the overall strategic planning and business development areas of the Company. Before assuming his present position, he was Corporate Staff Vice President of Marketing.

Mr. Rubin was elected a Vice President of the Company in May 1984. Prior to joining the Company he was associated with Atlantic Richfield Company, most recently as Vice President Planning & Control, ARCO Metals Company.

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. Prior to joining the Company, he held several executive positions in county government in Ventura, California.

Dr. Yon was elected a Vice President of the Company in April 1985. Dr. Yon joined C-E in January 1984 and was placed in charge of the Engineered Systems and Controls Group. Prior to joining C-E he was a consultant with Booz, Allen & Hamilton most recently as Vice President and lead partner serving high technology industries.

PART IV

ITEM 14 EXHIBITS AND FINANCIAL STATEMENT SCHEDULES

(a) 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) References to the Annual Report shall be deemed to refer to the Annual Report to Shareholders for the year ended December 31, 1985.

Schedules I, II, III, IV, IX, X, XI, XII and XIII are not submitted because they are not applicable or not required.

3. Exhibits —

(3) Restated Certificate of Incorporation of Combustion Engineering, Inc — Note (a)
By-Laws of Combustion Engineering, Inc — Note (b)

(10) 1982 Stock Option Plan of Combustion Engineering, Inc — Note (c)
Amended Incentive Compensation Plan, as amended November 21, 1985
Deferred Compensation Plan for Non-Employee Directors, as amended November 21, 1985
Deferred Compensation Plan
Key Employee Retention and Severance Benefit Plan and Form of Agreement — Note (d)
Executive Retirement and Life Insurance Plan, as amended November 21, 1985, and Form of Life Insurance Agreement
Supplemental Retirement Benefit Agreement with Charles E. Hugel — Note (d)
Supplemental Benefit Plan for Salaried Employees — Note (e)
Agreement with Arthur J. Santry, Jr — Note (g)
Consulting Agreement with Robert C. Seamans, Jr — Note (f)

(11) Computation of Net Income Per Share

(13) Annual Report to Shareholders for the year ended December 31, 1985

(22) Subsidiaries of the Registrant

(24) Consent of Experts

(25) Powers of Attorney

NOTES

(a) Incorporated by reference to Form 10-Q for the second quarter of 1983
(b) Incorporated by reference to Form 10-Q for the second quarter of 1985
(c) Incorporated by reference to Proxy Statement for Annual Meeting on April 27, 1982
(d) Incorporated by reference to Form 10-K Report for 1982
(e) Incorporated by reference to Form 10-K Report for 1983
(f) Incorporated by reference to Form 10-K Report for 1980 and 1981
(g) Incorporated by reference to Form 10-K Report for 1984

Report of Independent Public Accountants on Schedules

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

In connection with our examinations of the financial statements included in the Combustion Engineering, Inc. Annual Report to Shareholders and incorporated by reference in this Form 10-K, we have also examined the supplemental schedules listed in Item 14(a)2. Our examinations of the financial statements were made for the purpose of forming an opinion on those statements taken as a whole. The supplemental schedules are presented for purposes of complying with the Securities and Exchange Commission's rules and are not part of the basic financial statements. The supplemental schedules have been subjected to the auditing procedures applied in the examinations 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

Stamford, Connecticut
February 14, 1986

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES

PROPERTY, PLANT AND EQUIPMENT

YEARS ENDED DECEMBER 31, 1983, 1984 AND 1985

(Dollars in Thousands)

Column A	Column B	Column C		Column D	Column E	Column F
		Additions at Cost[3]			Other Changes	
Classification	Balance at Beginning of Period	Beginning Balance of Acquired Companies[1]	Other	Retirements[4]	Add (Deduct)	Balance at End of Period
Year Ended December 31, 1983:						
Land and land improvements	$ 49,305	$ 3,325	$ 1,402	$ 3,222	($ 150)	$ 50,880
Clay and other mineral deposits	34,377	—	3,514	—	—	37,891
Buildings	208,687	8,834	8,380	5,331	(743)	215,827
Machinery and equipment	606,533	22,836	35,504	23,720	(6,968)	634,185
Construction in progress	27,534	672	2,540[2]	—	(51)	30,695
	$ 928,436	$ 35,667	$ 49,340	$ 32,273	($ 9,912)[5]	$ 969,258
Year Ended December 31, 1984:						
Land and land improvements	$ 50,880	$ 1,108	$ 571	$ 7,652	($ 235)	$ 44,452
Clay and other mineral deposits	37,891	—	—	1,097	—	36,794
Buildings	215,827	8,916	2,943	17,905	(1,976)	207,805
Machinery and equipment	634,185	44,054	51,739	59,731	(2,713)	667,534
Construction in progress	30,695	935	(996)[2]	4	(9)	30,621
	$ 969,258	$ 55,013	$ 54,257	$ 88,389	($ 4,933)[5]	$ 987,206
Year Ended December 31, 1985:						
Land and land improvements	$ 44,452	$ 338	$ 1,271	$ 3,917	$ 35	$ 42,177
Clay and other mineral deposits	36,794	—	68	2,934	228	34,152
Buildings	207,805	2,249	6,307	14,234	1,986	204,113
Machinery and equipment	667,534	2,168	58,202	28,911	237	699,730
Construction in progress	30,621	—	11,960[2]	—	(4)	42,577
	$ 98[illegible],206	$ 4,753	$ 77,806	$ 49,998	$ [illegible]480[5]	$1,022,249

NOTES

[1] Includes $33,110 with respect to Taylor Instrument Company in 1983 and $39,232 and $9,150 with respect to Jamesbury Corp and Impell Corporation, respectively, in 1984

[2] Net of transfers to completed property, plant and equipment

[3] Reference is made to Note 1(g) of the Notes to Financial Statements on page 40 of the "Financial Section" of the Annual Report

[4] Includes $15,888 in 1983 relating to the sale of building products manufacturing and distribution facilities

[5] Balance at the beginning of 1983 has been restated for translation adjustments resulting from the adoption of FAS 52. The amounts shown in Column E above represent translation adjustments applicable to the years 1983, 1984 and 1985. Reference is made to Note 1(f) of the Notes to Financial Statements on page 39 of the "Financial Section" of the Annual Report.

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
ACCUMULATED DEPRECIATION, DEPLETION AND AMORTIZATION OF PROPERTY, PLANT AND EQUIPMENT

YEARS ENDED DECEMBER 31, 1983, 1984 AND 1985

(Dollars in Thousands)

Column A	Column B	Column C		Column D	Column E	Column F
		Additions			Other Changes	
Classification	Balance at Beginning of Period	Beginning Balance of Acquired Companies	Charged to Costs and Expenses	Retirements[2]	Add (Deduct)	Balance at End of Period
Year Ended December 31, 1983:						
Land improvements	$ 6,335	$ —	$ 1,714	$ 48	$ 7	$ 8,008
Clay and other mineral deposits	2,277	—	3,272	—	—	5,548
Buildings	94,414	—	15,471	1,928	1	107,958
Machinery and equipment	348,271	—	55,616	15,282	(4,481)	384,144
	$451,297	$ —	$ 78,073[1]	$ 17,258	($ 4,453)[3]	$505,859
Year Ended December 31, 1984:						
Land improvements	$ 8,008	$ —	$ 1,369	$ 582	($ 12)	$ 8,803
Clay and other mineral deposits	5,549	—	365	453	—	5,461
Buildings	107,958	—	3,806	4,702	(296)	106,766
Machinery and equipment	384,144	—	45,136	38,487	(1,512)	381,281
	$505,859	$ —	$ 50,676	$ 42,[illegible]04	($ 1,820)[3]	$512,311
Year Ended December 31, 1985:						
Land improvements	$ 8,803	$ —	$ 558	$ 492	($ 12)	$ 8,857
Clay and other mineral deposits	5,461	—	456	489	—	5,478
Buildings	106,766	—	1,522	9,065	(28)	99,187
Machinery and equipment	391,281	—	72,293	38,475	(116)	424,983
	$512,311	$ —	$ 74,829[1]	$ 48,521	($ 154)[3]	$538,465

NOTES

[1] Includes $20,585 and $12,300 representing adjustments to certain fixed assets in connection with the provisions for adjustments to facilities and operations in 1983 and 1985, respectively. Reference is made to Note 3 of the Notes to Financial Statements on page 42 of the "Financial Section" of the Annual Report.

[2] Includes $5,996 in 1983 relating to the sale of building products manufacturing and distribution facilities.

[3] Balance at beginning of period has been restated for translation adjustments resulting from the adoption of FAS 52. The amount shown in Column E above represents translation adjustments applicable to the years 1983, 1984 and 1985. Reference is made to Note 1(f) of the Notes to Financial Statements on page 39 of the "Financial Section" of the Annual Report.

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES

GUARANTEES OF SECURITIES OF OTHER ISSUERS

DECEMBER 31, 1985

(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
Lummus Crest Inc./ Salzgitter Lummus G.m.b.H. Joint Venture	Overdraft Facility	$ 2,001	$ —	$ —	(1)	None
Jamesbury Corp./ OyM S.A.	Bank Note	927	—	—	(1)	None
Jamesbury Corp./ Hammel Dahl, Inc.	Note	1,000	—	—	(1)	None
		$ 3,928	$ —	$ —		

NOTE

(1) Guarantee of principal and interest.

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
VALUATION AND QUALIFYING ACCOUNTS AND RESERVES
YEARS ENDED DECEMBER 31, 1983, 1984 AND 1985
(Dollars in Thousands)

Column A	Column B	Column C			Column D	Column E
		Additions				
Description	Balance at Beginning of Period	Beginning Balance of Acquired Companies[1]	Charged to Costs and Expenses	Charged to Other Accounts	Deductions	Balance at End of Period
Year Ended December 31, 1983:						
Reserves Deducted In The Balance Sheet From The Asset To Which They Apply:						
Reserve for doubtful accounts and allowances	$11,738	$ 740	$ 6,230	$ —	$ 2,854[2]	$15,854
Reserves Included in Current Liabilities:						
Reserve for warranty[1]	$36,009	$ 845	$ 7,109	$ —	$ 4,379[3]	$39,584
Reserve for supplementary pension plan	2,405	—	813	—	260[4]	2,958
	$38,414	$ 845	$ 7,922	$ —	$ 4,639	$42,542
Year Ended December 31, 1984:						
Reserves Deducted In The Balance Sheet From The Asset To Which They Apply:						
Reserve for doubtful accounts and allowances	$15,854	$ 222	$ 3,081	$ —	$ 4,114[2]	$15,043
Reserves Included In Current Liabilities:						
Reserve for warranty[1]	$39,584	$ —	$ 3,898	$ —	$ 4,978[3]	$38,504
Reserve for supplementary pension plan	2,958	—	520	—	101[4]	3,377
	$42,542	$ —	$ 4,418	$ —	$ 5,079	$41,881

Schedule VIII continued on following page.

SCHEDULE VIII
Continued

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES

VALUATION AND QUALIFYING ACCOUNTS AND RESERVES

YEARS ENDED DECEMBER 31, 1983, 1984 AND 1985

(Dollars in Thousands)

Column A	Column B	Column C			Column D	Column E
		Additions				
Description	Balance at Beginning of Period	Beginning Balance of Acquired Companies[1]	Charged to Costs and Expenses	Charged to Other Accounts	Deductions	Balance at End of Period
Year Ended December 31, 1985:						
Reserves Deducted In The Balance Sheet From The Asset To Which They Apply:						
Reserve for doubtful accounts and allowances	$15,043	$ —	$ 3,443	$ —	$ 1,895[2]	$16,591
Reserves Included In Current Liabilities:						
Reserve for warranty[5]	$38,504	$ —	$18,042	$ —	$ 9,427[3]	$47,119
Reserve for supplementary pension plan	3,377	—	750	—	351[4]	3,776
	$41,881	$ —	$18,792	$ —	$ 9,778	$50,895

NOTES

[1]Represents reserve accounts of acquired companies

[2]Represents uncollectible receivables

[3]Represents additional costs incurred and adjustments

[4]Represents supplemental pension payments

[5]See comment in Item 1(b) on page 2 with respect to cost estimates for long-term contracts and provisions for future warranty costs

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 Charles E. Hugel
President and Chief Executive Officer, Director

By George S. Kimmel
Executive Vice President and Chief Financial Officer, Director

By Jeffrey S. Rubin
Vice President and Controller

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.

By Arthur J. Santry, Jr.
Chairman of the Board

By Harry J. Bolwell*
Director

By Thomas A. Ennis*
Director

By Walter H. Helmerich, III
Director

By Robert M. Jenney
Director

By ______________________
Paul W. MacAvoy
Director

By Dudley C. Mecum
Director

By Scott L. Probasco, Jr.
Director

By Robert C. Seamans, Jr.
Director

By Robert G. Stone, Jr.
Director

By Kenneth J. Whalen
Director

*Pursuant to Power of Attorney

Dated March 27, 1986

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, 1981 — Commission file number 1-117-2

COMBUSTION ENGINEERING, INC.

(Exact Name of Registrant As Specified in its Charter)

Delaware	13-1587569
(State or Other Jurisdiction of Incorporation or Organization)	(I.R.S. Employer Identification No.)
900 Long Ridge Road, Stamford, Connecticut	06902
(Address of Principal Executive Offices)	(Zip Code)

Registrant's telephone number, including area code (203) 329-8771

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:

(Title of Class)

(Title of Class)

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 12, 1982
Common Stock — $1 par value	33,016,162

The aggregate market value of the voting stock held by non-affiliates of the registrant on March 12, 1982, was approximately $834,000,000.

Documents Incorporated By Reference

The following documents are incorporated by reference:

(1) "Financial Section" of the Annual Report to Shareholders for the year ended December 31, 1981, in response to Items 1(a) and 1(d) and Item 3 of Part I, and Items 5 through 8 of Part II; and in partial response to Items 1(b) and 1(c) of Part I, and Item 11 of Part IV.

(2) The Company's proxy statement dated March 26, 1982, in connection with its Annual Meeting of Stockholders to be held on April 27, 1982, in response to Item 4 of Part I; and Items 9 and 10 of Part III.

PART I

ITEM 1. DESCRIPTION OF THE BUSINESS

References to the Company contained herein shall be deemed to refer to the Company and its consolidated subsidiaries. References to the Annual Report shall be deemed to refer to the Annual Report to Shareholders for the year ended December 31, 1981.

Item 1(a) General Development of Business

Reference is made to Note 2 of the Notes to Financial Statements on page 33 of the "Financial Section" of the Annual Report regarding acquisitions and dispositions.

Item 1(b) Financial Information About Industry Segments

Reference is made to Note 13 of the Notes to Financial Statements on pages 37 to 39 of the "Financial Section" of the Annual Report regarding financial reporting by business segments.

Much of the Company's business, especially that relating to steam generating systems, equipment and services for the electric utility industry and design, engineering and construction services for the chemical, petrochemical and petroleum industries, involves long-term contracts of various types, including fixed price and cost plus fee type contracts with some contracts including variations of both types. Certain contracts include incentive provisions whereby the profit is adjusted depending on performance. The largest proportion 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 recorded principally on the basis of the estimated stage of completion. However, 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. These contracts extend over a period of from several months to four or more years. Revisions in cost estimates during the progress of the work under long-term contracts have the effect of including in subsequent accounting periods adjustments necessary to reflect the results indicated by the revised estimates of final cost. Projected or realized losses under long-term contracts, if any, are provided for in the period when first determined. See Note 1(d) of Notes to Financial Statements on page 30 of the "Financial Section" of the Annual Report.

Cost estimates for long-term contracts take into account all anticipated costs, including, among others, engineering, manufacturing, subcontracting and field construction costs which are required to meet the specifications, including warranties, of the contracts. In addition, when a long-term contract for steam generating equipment is completed for accounting purposes (usually after payment by the customer of amounts retained under terms of the contract and satisfactory operating performance of the equipment), provision is made for future warranty costs, generally on the basis of past experience.

Item 1(c) Narrative Description of the Business

Reference is made to "Business Segments and Brief Description of the Business" shown on pages 40 and 41 of the "Financial Section" of the Annual Report regarding a narrative description of the Company's business.

Raw Materials

The principal 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.

The uncertain availability of natural gas in past years required the Company to develop alternative sources of energy for certain of its operations. Substitute forms of energy, while available, are in most cases more costly than natural gas.

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 the business. 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.

Reference is made to the "Management Discussion and Analysis of Financial Condition and Results of Operations — Unfilled Orders" on page 25 of the "Financial Section" of the Annual Report. Approximately 58% of the consolidated December 31, 1981, backlog of unfilled orders is expected to be recorded as sales (principally on the percentage of completion method) in 1982 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 equipment, products and services for the oil and gas market, the Company, in certain cases, is one of the leading manufacturers or suppliers and competes with a number of companies including Cameron Iron Works; National Supply Company, a division of ARMCO; Smith Industries, Inc.; Sivalls, Inc.; Maloney Crawford Tank Co. and Wellhead Equipment Division of FMC Corporation. With respect to other equipment, products and services for industrial markets, 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.

With respect to steam generating systems, equipment and services for the electric utility industry, the Company is one of the largest domestic manufacturers of fossil fueled steam generating systems and equipment and is one of four domestic manufacturers of nuclear steam supply systems. The competitors for fossil fueled steam generating systems include The Babcock & Wilcox Company, a wholly-owned subsidiary of McDermott Incorporated and Foster Wheeler Corporation. The competitors for nuclear steam supply systems are The Babcock & Wilcox Company, Westinghouse Electric Corporation and General Electric Company.

The Lummus Company, the principal component of the design, engineering and construction services segment of the Company, is one of the ten largest domestic firms engaged in designing, engineering and constructing chemical process plants, petroleum refineries and other industrial facilities for the petrochemical, metallurgical, pulp and paper and other process industries. The principal competitors for this business include Bechtel Corporation, Brown and Root, Inc., a subsidiary of Halliburton Company, Stone and Webster Engineering Corporation, Fluor Corporation, Foster Wheeler Corporation and The M. W. Kellogg Co., a wholly-owned subsidiary of Wheelabrator-Frye Inc.

Usually, the Company competes for new orders for fossil fueled steam generating systems and equipment and for nuclear steam supply systems by responding to specific invitations to bid. This same process is usually involved in securing orders for the design, engineering and construction of chemical process plants and other plants sold by The Lummus Company. The principal methods of competition would include the following factors, but not necessarily in the order of importance: design of the equipment or process to be furnished in response to the customer's specifications, technical support and service, ability to meet the customer's delivery schedule and price.

Research and Development

The estimated amount spent during 1981, 1980 and 1979 on material research activities relating to the development of new products or services or the improvement of existing products or services which was Company sponsored was $47,340,000, $40,000,000 and $37,700,000, respectively, and on that which was customer sponsored was $35,500,000, $37,400,000, and $24,500,000, respectively.

Compliance with Environmental Protection Laws

Compliance by the Company with Federal, state and local environmental protection laws required capital expenditures of $3,990,000 in 1981, $2,745,000 in 1980 and $2,547,000 in 1979. It is estimated that capital expenditures in 1982 for such purpose will be at approximately the same level as in 1981.

Employees

At December 31, 1981, the Company employed 46,704 persons.

Item 1(d) Financial Information about Foreign and Domestic Operations and Export Sales

Reference is made to Note 13 of the Notes to Financial Statements shown on pages 37 to 39 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 used the property is also identified. Unless noted, the property is owned by the Company or a subsidiary.

Aberdeen, Scotland (1)
Andersonville, Georgia (2)*#
Bakersfield, California (2)
Beaumont, Texas (2)
Birmingham, Alabama (1) (3)
Bloomfield, New Jersey (4)
Brantford, Ontario (1)
Calgary, Alberta (1)
Celle, West Germany (1)
Chattanooga, Tennessee (1) (2) (3)
Cleveland, Ohio (2)*
Douglas, Scotland (1)*
Dry Branch, Georgia (2)#
East Chicago, Indiana (1) (3)
Enterprise, Kansas (1)
Gabbs, Nevada (2)*#
Gulfport, Mississippi (1)*
Houston, Texas (1) (4)*
London, England (4)*
Mansfield, Texas (1)
Maple Grove, Ohio (2)*#
Marion, North Carolina (1) (3)
Mentor, Ohio (1) (2)
Monongahela, Pennsylvania (1) (3)
Newington, New Hampshire (3)
Newell, West Virginia (2)#
Oshkosh, Wisconsin (2)
Pasadena, Texas (2)
Sandersville, Georgia (2)#
Savannah, Georgia (2)
Sherbrooke, Quebec (1) (3)
Springfield, Ohio (1)*
Stamford, Connecticut (Corporate Office)
St. Catherines, Ontario (1) (2)
St. Louis, Missouri (1) (3)
The Hague, Netherlands (4)*
Tulsa, Oklahoma (1) (2)*
Ventura, California (1)
Wellsville, New York (1) (3)
Windsor, Connecticut (1) (3)

(1) Equipment for oil and gas and other industrial markets.
(2) Products and services for oil and gas and other industrial markets.
(3) Steam generating systems, equipment and services for the electric utility industry.
(4) Design, engineering and construction services.

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 11 of the Notes to Financial Statements shown on page 37 of the "Financial Section" of the Annual Report for the year ended December 31, 1981.

ITEM 4. SECURITY OWNERSHIP OF CERTAIN BENEFICIAL OWNERS AND MANAGEMENT

The Company's proxy statement dated March 26, 1982, in connection with its Annual Meeting of Stockholders to be held on April 27, 1982, has been filed with the Securities and Exchange Commission and the information with respect to stock ownership set forth on pages 2 and 3 thereof is hereby incorporated by reference.

OFFICERS OF THE REGISTRANT

Listed below are the officers of the Company:

Name	Age	Position Presently Held
Arthur J. Santry, Jr.	63	President
James F. Calvert	61	Vice President-Operations
Joseph F. Condon	56	Vice President-International
William J. Connolly	52	Vice President-Corporate and Investor Relations
Thomas A. Ennis	61	Vice President-Administration
Richard J. Hallinan	59	Vice President, Secretary and General Counsel
James B. Kelly	54	Vice President in charge of Industrial Products Group
Mitchell Kiamie	60	Vice President and Controller
George S. Kimmel	47	Vice President-Finance
Sven A. Kreipke	56	Vice President in charge of Engineering and Construction Group
Charles K. Leeper	58	Vice President-Corporate Technology
John F. Mangold	55	Vice President in charge of Process Equipment Group
Robert H. Masson	46	Vice President and Treasurer
William P. Orr	64	Vice President
John R. Peterson	48	Vice President-Planning and Business Development
John H. Slack	59	Vice President in charge of Oil and Gas Group
Gene M. Wilkinson	52	Vice President-Personnel Administration
Howard M. Winterson	65	Vice President in charge of Power Systems Group

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. Santry was elected President in April, 1983. During the past five years he has been the chief executive officer of the Company. He has been a Director since 1967.

Mr. Calvert was elected a Vice President of the Company in August, 1974. During the past five years he has been the Vice President-Operations. He was elected a Director in February, 1975.

Mr. Condon was elected a Vice President of the Company in January, 1978 and is responsible for coordinating the Company's international manufacturing and licensing activities, sales and market services, and project financing outside the United States. Before assuming his present position, he was Vice President-International Finance for The Lummus Company, a subsidiary of the Company.

Mr. Connolly was elected a Vice President of the Company in April, 1975 and was responsible for corporate marketing and communications until September, 1980 when he became Vice President-Corporate and Investor Relations.

Mr. Ennis was elected a Vice President of the Company in January, 1960. During the past five years he has been responsible for the Company's legal and administrative functions. He has been a Director since 1960.

Mr. Hallinan was elected a Vice President of the Company in November, 1975 and Secretary in April, 1973. He became General Counsel in 1973 and since that time has been in charge of the legal department of the Company.

Mr. Kelly was elected a Vice President of the Company in April, 1967. During the past five years he has been a senior operating officer of the Industrial Products Group.

Mr. Klamas was elected a Vice President of the Company in August, 1967. During the past five years he has been the principal accounting officer of the Company.

Mr. Kimmel was elected a Vice President of the Company in April, 1979 and has been Vice President-Finance since June, 1980. He was elected a Director in April, 1981. Prior to joining the Company, he was Vice President-Finance and a member of the Board of Directors of Lykes Corporation.

Mr. Kreipke was elected a Vice President of the Company in June, 1981. He is a senior operating officer of the Engineering and Construction Group. Before assuming his present position, he was President of The Lummus Company, a subsidiary of the Company, and prior to that, a Vice President of that company.

Mr. Leeper was elected a Vice President of the Company in December, 1978 and is responsible for directing corporate-level technology programs, and for coordination of the Company's engineering and technological development. Before assuming his present position, he was Vice President-Engineering at The Air Preheater Company, Inc., a subsidiary of the Company. Prior to joining the Company, he was President and General Manager of Aerojet Nuclear Company, a subsidiary of Aerojet General Corporation.

Mr. Mangold was elected a Vice President of the Company in January, 1982. Effective with his election he was placed in charge of the Process Equipment Group. Prior to assuming his present position, he was President of Vetco Offshore, Inc., a subsidiary of the Company, and prior to that, Vice President of Nuclear Manufacturing for the Power Systems Group.

Mr. Masson was elected a Vice President of the Company in November, 1980. Prior to joining the Company, he was Vice President and Treasurer of PepsiCo, Inc. and, prior to that, Vice President-Finance and Treasurer of Ellerbe, Inc.

Mr. Orr was elected a Vice President of the Company in September, 1973. During most of the past five years he was the senior operating officer of the Engineering Group. He ceased being the senior operating officer in June, 1981 and will retire in August, 1982.

Mr. Peterson was elected a Vice President of the Company in September, 1980 and is responsible for directing corporate-level programs in the overall strategic planning and business development areas of the Company. Before assuming his present position, he was Corporate Staff Vice President of Marketing. Prior to joining the Company, he was Manager of Corporate Marketing and Strategic Planning, Consulting Operation for General Electric Company.

Mr. Slack was elected a Vice President of the Company in September, 1973. During the past five years he has been a senior operating officer of the Process Equipment Group. Effective in January, 1982 he was placed in charge of the Oil and Gas Group.

Mr. Wilkinson was elected a Vice President of the Company in June, 1981 and is responsible for the Corporate Personnel function which includes management training and development programs, organization planning and staffing, compensation, benefits, security, personnel research activities and overall personnel policies. Prior to joining the Company, he was Assistant to the Chairman and Chief Executive Officer of Baxter Travenol Laboratories, Inc.

Mr. Winterson was elected a Vice President of the Company in December, 1966. During the past five years he has been a senior operating officer of the Power Systems Group. He has been a Director since 1968.

PART II

ITEMS 5. THROUGH 8.

The "Financial Section" of the Annual Report to Shareholders for the year ended December 31, 1981, 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	46 to 48
Item 6 — Selected Financial Data	18
Item 7 — Management Discussion and Analysis of Financial Condition and Results of Operations	19 to 25
Item 8 — Financial Statements and Supplementary Data	26 to 39 and 42 to 45

PART III

ITEMS 9. AND 10.

The Company's proxy statement dated March 26, 1982, in connection with its Annual Meeting of Stockholders to be held on April 27, 1982, has been filed with the Securities and Exchange Commission and the information set forth under "Information on Nominees for Directors" on pages 2 and 3 thereof and under "Management Remuneration and Transactions" on pages 4 to 8 thereof is hereby incorporated by reference. Reference is also made to "Supplement to Part I — Officers of the Registrant" set forth on pages 5 and 6.

PART IV

ITEM 11. EXHIBITS AND FINANCIAL STATEMENT SCHEDULES

(a) Documents:

NOTES:

(a) Separate financial statements of the registrant have been omitted since it is primarily an operating company and the minority interests in subsidiaries and long-term debt of the subsidiaries held by others than the registrant is less than five percent of consolidated total assets.

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

(c) References to the Annual Report shall be deemed to refer to the Annual Report to Shareholders for the year ended December 31, 1981.

Schedules I, II, III, IV, IX, X, XI, XII and XIII are not submitted because they are not applicable or not required.

3. Exhibits —

(3) Restated Certificate of Incorporation of Combustion Engineering, Inc.*

By-Laws of Combustion Engineering, Inc.*

(10) Amendment and Restatement of Combustion Engineering, Inc. 1970 Stock Option Plan*

December 30, 1981, Amendment to Amendment and Restatement of Combustion Engineering, Inc. 1970 Stock Option Plan

Amended Incentive Compensation Plan*

Deferred Compensation Plan for Non-Employee Directors*

Supplemental Benefit Plan for Salaried Employees*

Consulting Agreement with Robert C. Seamans, Jr.*

Amendment to Consulting Agreement with Robert C. Seamans, Jr.

(11) Computation of Net Income Per Share

(13) Annual Report to Shareholders for the year ended December 31, 1981

(22) Subsidiaries of the Registrant

*Reference is made to the Company's annual report on Form 10-K for the year ended December 31, 1980.

Report of Independent Public Accountants on Schedules

In connection with our examinations of the financial statements included in the Combustion Engineering, Inc. Annual Report to Shareholders and incorporated by reference in this Form 10-K, we have also examined the supplemental schedules listed in Item 11(a)2. Our examinations were made for the purpose of forming an opinion on the basic financial statements taken as a whole. The supplemental schedules are presented for purposes of complying with the Securities and Exchange Commission's rules and regulations under the Securities and Exchange Act of 1934 and are not otherwise a required part of the basic financial statements. The supplemental schedules have been subjected to the auditing procedures applied in the examinations 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.

Stamford, Connecticut
February 11, 1982

SCHEDULE

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES

PROPERTY, PLANT AND EQUIPMENT

YEARS ENDED DECEMBER 31, 1979, 1980 AND 1981

(Dollars in Thousands)

Column A	Column B	Column C		Column D	Column E	Column F
		Additions at Cost[1]			Other Changes	
Classification	Balance at Beginning of Period	Beginning Balance of Acquired Companies[1]	Other	Retirements	Add (Deduct)	Balance at End of Period
Year Ended December 31, 1979:						
Land and land improvements	$ 30,878	$ 4,381	$ 2,886	$ 350	$ —	$ 37,503
Clay and other mineral deposits	1,837	3,877	299	—	—	8,013
Buildings	209,621	11,692	11,826	4,811	—	228,128
Machinery and equipment	821,694	38,409	88,514	30,444	—	588,173
Construction in progress	22,181	611	18,637[2]	—	—	39,438
	$ 786,118	$ 88,870	$ 87,872	$ 35,605	$ —	$ 897,258
Year Ended December 31, 1980:						
Land and land improvements	$ 37,603	$ 12,968	$ 8,412	$ 121	$ —	$ 55,780
Clay and other mineral deposits	8,013	31,590	38	51	—	37,587
Buildings	228,128	3,878	8,665	3,943	—	238,928
Machinery and equipment	588,173	26,112	108,371	12,435	—	706,221
Construction in progress	39,438	17,570	19,346[2]	—	—	78,355
	$ 897,258	$ 92,114	$140,029	$ 16,550	$ —	$1,112,849
Year Ended December 31, 1981:						
Land and land improvements	$ 58,780	$ 801	$ 14,911	$ 4,685	$ —	$ 68,807
Clay and other mineral deposits	37,587	—	426	8,453	—	31,560
Buildings	238,928	3,859	21,185	15,481	—	248,509
Machinery and equipment	706,221	2,220	138,174	92,604	—	752,111
Construction in progress	78,355	305	728[2]	4,584	—	72,801
	$1,112,849	$ 6,985	$173,421	$123,687[4]	$ —	$1,169,588

NOTES:

[1] Represents the property, plant and equipment of acquired companies. Reference is made to Note 2 of Notes to Financial Statements on page 33 of the "Financial Section" of the Annual Report.

[2] Net of transfers to completed property, plant and equipment.

[3] Reference is made to Note 1(g) of Notes to Financial Statements on page 31 of the "Financial Section" of the Annual Report.

[4] Includes $86,238 relating to the sale of the Company's glass manufacturing ...

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
ACCUMULATED DEPRECIATION, DEPLETION AND AMORTIZATION OF PROPERTY, PLANT AND EQUIPMENT
YEARS ENDED DECEMBER 31, 1979, 1980 AND 1981

(Dollars in Thousands)

Column A	Column B	Column C		Column D	Column E	Column F
		Additions			Other Changes	
Description	Balance at Beginning of Period	Beginning Balance of Acquired Companies	Charged to Costs and Expenses	Retirements	Add (Deduct)	Balance at End of Period
Year Ended December 31, 1979:						
Land improvements	$ 3,913	$ —	$ 680	$ 75	$ —	$ 4,518
Clay and other mineral deposits	384	—	229	—	—	613
Buildings	73,235	—	10,496	2,184	—	81,547
Machinery and equipment	257,720	—	54,733	17,859	—	294,594
	$336,252	$ —	$ 66,138	$ 20,118	$ —	$381,272
Year Ended December 31, 1980:						
Land improvements	$ 4,518	$ —	$ 703	$ 25	$ —	$ 5,196
Clay and other mineral deposits	613	—	430	—	—	1,043
Buildings	81,547	—	12,958	376	—	94,129
Machinery and equipment	294,594	—	68,179	10,858	—	351,915
	$381,272	$ —	$ 82,270	$ 11,259	$ —	$452,283
Year Ended December 31, 1981:						
Land improvements	$ 5,196	$ —	$ 1,321	$ 105	$ —	$ 6,412
Clay and other mineral deposits	1,043	—	697	2	—	1,738
Buildings	94,129	—	14,083	5,687	—	102,525
Machinery and equipment	351,915	—	82,723	46,757	—	387,881
	$452,283	$ —	$ 98,824	$ 52,551[1]	$ —	$498,656

NOTE:

[1] Includes $40,322 relating to the sale of the Company's glass manufacturing and processing plants.

SCHEDULE

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
GUARANTEES OF SECURITIES OF OTHER ISSUERS [1]

DECEMBER 31, 1981

(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
Equipetrol — Industria E Comercio, S.A.	Bank Loan	$ 87	$ —	$ —	[2]	None
The Lummus Company/ Thyssen Rheinstahl Technik, G.m.b.H. Joint Venture	Overdraft Facility	8,112	—	—	[2]	None
Salzgitter Lummus G.m.b.H.	Bank Overdraft	545	—	—	[2]	None
		$ 8,744	$ —	$ —		

NOTES:

[1] The foregoing data . . cludes discounted notes receivable

[2] Guarantee of principal and interest.

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
VALUATION AND QUALIFYING ACCOUNTS AND RESERVES
YEARS ENDED DECEMBER 31, 1979, 1980 AND 1981
(Dollars In Thousands)

Column A	Column B	Column C			Column D	Column E
		Additions				
Description	Balance at Beginning of Period	Beginning Balance of Acquired Companies [1]	Charged to Costs and Expenses	Charged to Other Accounts [2]	Deductions	Balance at End of Period
Year Ended December 31, 1979:						
Reserves Deducted In The Balance Sheet From The Asset To Which They Apply:						
Reserve for doubtful accounts and allowances	$14,332	$ 133	$ 6,793	$ —	$ 6,429 [3]	$13,829
Reserves Included In Current Liabilities:						
Reserve for additional costs and possible future expenses on completed contracts [4]	$13,681	$ 276	$10,579	$ 2,323	$ 3,948 [5]	$22,711
Reserve for supplementary pension plan	1,699	—	91	—	197 [6]	1,663
	$15,380	$ 276	$10,670	$ 2,323	$ 4,145	$24,374
Year Ended December 31, 1980:						
Reserves Deducted In The Balance Sheet From The Asset To Which They Apply:						
Reserve for doubtful accounts and allowances	$13,829	$ 144	$ 6,890	$ —	$ 6,502 [3]	$14,361
Reserves Included In Current Liabilities:						
Reserve for additional costs and possible future expenses on completed contracts [4]	$22,711	$ —	$12,393	$ 1,779	$ 8,605 [5]	$28,278
Reserve for supplementary pension plan	1,663	—	640	312	123 [6]	2,492
	$24,374	$ —	$13,033	$ 2,091	$ 8,728	$30,770

Schedule VIII continued on following page.

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COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
VALUATION AND QUALIFYING ACCOUNTS AND RESERVES
YEARS ENDED DECEMBER 31, 1979, 1980 AND 1981
(Dollars in Thousands)

Column A	Column B	Column C			Column D	Column E
		Additions				
Description	Balance at Beginning of Period	Beginning Balance of Acquired Companies[1]	Charged to Costs and Expenses	Charged to Other Accounts[6]	Deductions	Balance at End of Period
Year Ended December 31, 1981:						
Reserves Deducted In The Balance Sheet From The Asset To Which They Apply:						
Reserve for doubtful accounts and allowances	$14,381	$ 102	$ 4,707	$ —	$ 3,630[2]	$18,840
Reserves Included in Current Liabilities:						
Reserve for additional costs and possible future expenses on completed contracts[5]	$28,278	$ 387	$20,117	$ 3,657	$ 9,020[3]	$43,288
Reserve for supplementary pension plan	2,492	—	82	—	264[4]	2,320
	$30,770	$ 387	$20,209	$ 3,657	$ 9,284	$45,608

NOTES:

[1] Represents reserve accounts of acquired companies. Reference is made to Note 2 of Notes to Financial Statements on page 33 of the "Financial Section" of the Annual Report.

[2] Represents uncollectible receivables and sales allowances granted.

[3] Represents additional costs incurred and adjustments.

[4] Represents supplemental pension payments.

[5] See , ment in Item 1(b) on page 2 with respect to cost estimates for long-term contracts and provisions for future warranty costs.

[6] Reclassified from accrued liabilities.

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 Arthur J. Santry, Jr.
President and Director

By George S. Kimmel
Vice President-Finance and Director

By Mitchell Kismie
Vice President and Controller

report has been signed below by the following persons on behalf of the registrant and in the capacities and on the date indicated.

By James F. Calvert
Director

By W. Van Alan Clark, Jr.
Director

By Thomas A. Ennis
Director

By Walter H. Helmerich, III
Director

By Robert M. Jenney
Director

By Paul W. MacAvoy
Director

By Scott L. Probasco, Jr.
Director

By Robert C. Seamans, Jr.
Director

By Robert G. Stone, Jr.
Director

By James F. Thornton
Director

By Howard M. Winterson
Director

Dated: March 25, 1982

APPENDIX C

Validation of Criticality Methods for Calculating the Effective Multiplication Factor Under Low Hydrogen Density Moderation Conditions

By
Mohamed A. Elmaghrabi
Consulting Physicist

Approved by: R. J. Klotz
Principal Consulting Scientist

July 24, 1987

SUMMARY

Critical experiments for the interposition of hydrogenous compounds between four assemblies of 18 x 18 U(4.75%) O_2 rods were reported in Reference 1. The purpose of this report is to define and validate a detailed calculational model which would be suitable for MONTE CARLO calculations to design and license fuel storage under mist (low hydrogen density) conditions.

Results of this analysis are as follows:

1. A calculational model based on a homogenized fuel assembly representation and 16 neutron energy groups which is suitable for use in KENO analyses to design and license fuel storage under mist conditions has been defined and verified against critical experiments.

2. This model yields multiplication factors for critical experiments which are comparable to those obtained by other methods (APPOLLO-MORET) used in Reference 1.

ABSTRACT

The objective of this task is the development and validation of an analytical model for use in the design and licensing of Fuel Storage under mist conditions. Combustion Engineering, Inc. has performed analyses using the KENO computer code, which show that neutron multiplication factors obtained for the case of interposition of low hydrogen density compounds between four 18 x 18 fuel assemblies, are in good agreement with other calculations models and the experiments. The bias between prediction and measurement is reasonable and quantifiable over the range of experimental parameters.

July 24, 1987

CONTENTS

July 24, 1987

Section I

INTRODUCTION

This task has a its primary objective the development and validation of an analytical model for use in the criticality analysis of fuel storage under mist (low hydrogen density) conditions.

Experimental criticality data were measured on four assemblies of 18 x 18 (U4.75%) O_2 rods at 13.5 mm square pitch with variable low hydrogen density compounds interposed between them. The experimental data and subsequent analyses of these experiments were reported in Reference 1. The analyses were carried out using the APPOLLO-MORET computer codes in 16 neutron energy groups. It was the objective of this task to define an analytical model for design and licensing of fuel storage under mist conditions.

Section 2

BENCHMARKS

Critical experiments on the interposition of low hydrogen density materials between four assemblies of 18 x 18 U(4.75%) O_2 rods at 13.5 mm square pitch, were performed by the department of Nuclear Safety of the French Atomic Energy Commission (Reference 1). The assemblies were arranged in a 2 x 2 array in the experimental tank. A mobile device enabled them to move along the x and y axes in the horizontal plane (Figure 1).

Cross-shaped watertight containers in various thickness (25.5 to 100 mm) with 3 mm thick aluminum walls are then interpositioned between the four assemblies and successively filled with air and various hydrogenous compounds which are:

1. Air
2. Expanded polystyrene $(C_8H_8)_n$
3. Polyethylene power $(CH_2)_n$
4. Polyethylene balls $(CH_2)_n$
5. Water

Water is then introduced into the bottom of the tank and the critical height determined for each configuration. The resulting values are summarized in Table 1.

The calculated results shown in Table 1 were performed by the criticality service of the French Atomic Energy Commission. Two-step calculations were performed using the APPOLLO and MORET codes:

1. Calculation of the neutron constant by APPOLLO.
2. Calculation of the K_{eff} by MORET.

The APPOLLO code is used with the "transport" option to calculate the material buckling B^2 and the K_{eff} of an infinite lattice of rods in its ambient medium (water or air) and to determine the macroscopic cross-section of the homogenized lattice based on the 16 energy groups of Hansen and Roach. The cross-section library used in the code is ENDF/B-III based and consists of 99 groups (52 fast and 47 thermal groups).

The MORET code is a Monte Carlo code that calculates the K_{eff} of any configuration. The collisions are treated isotropically but aniso-tropy is taken into account by means of the transport corrections.

Results of Experiments and Benchmark Calculations in the Case of Interposition of Hydrogenous Compounds Between Four Assemblies of 18 × 18 U(4.75%) O_2 Rods at 13.5mm Square Pitch

	Experimental Results					Calculated Results
	Compounds					
Λ^{a} (cm)		Nature	Density (g/cm^3)	Concentration Hydrogen (g/cm^3)	Water Critical Height (mm)	APOLLO-MORET k_{eff} ±2σ
0	1.	Water	1.0	0.1119	238 ± 0.6	1.010 ± 0.011
2.5	2.	Box + air	0	0	290.3 ± 0.9	1.005 ± 0.010
	3.	Box + $(C_8H_8)n$	0.0323	0.0025	286.1 ± 0.8	0.987 ± 0.010
	4.	Box + powder $(CH_2)n$	0.2879	0.0414	269.8 ± 0.6	
	5.	Box + balls $(CH_2)n$	0.5540	0.0800	255.4 ± 0.6	0.995 ± 0.010
	6.	Box + water	1.0	0.1119	256.6 ± 0.7	
	7.	Water	1.0	0.1119	244.8 ± 0.6	1.006 ± 0.011
5.0	8.	Box + air	0	0	344.8 ± 0.7	
	9.	Box + $(C_8H_8)n$	0.0262	0.0020	343.9 ± 0.8	
	10.	Box + powder $(CH_2)n$	0.3335	0.0480	301.6 ± 0.6	
	11.	Box + balls $(CH_2)n$	0.5796	0.0833	307.3 ± 0.8	
	12.	Box + water	1.0	0.1119	327.8 ± 0.8	
	13.	Water	1.0	0.1119	314.7 ± 0.6	1.000 ± 0.012
10.0	14.	Box + air	0	0	460.8 ± 0.7	0.996 ± 0.010
	15.	Box + $(C_8H_8)n$	0.0288	0.0022	456.2 ± 0.8	0.987 ± 0.010
	16.	Box + powder $(CH_2)n$	0.3216	0.0464	420.5 ± 0.6	1.011 ± 0.012
	17.	Box + balls $(CH_2)n$	0.5680	0.0816	499.4 ± 0.6	1.010 ± 0.010
	18.	Box + water	1.0	0.1119	641.2 ± 0.9	
	19.	Water	1.0	0.1119	643.4 ± 0.8	

[a] The symbol Λ is the value of the gap width between the assemblies, thus it is the value of cross-shaped box width. The actual thickness of hydrogenous compounds is Λ (H) = Δ − 0.6 cm.

Section 3

MODEL EVALUATION

3.1 Computer Codes

The computer codes employed in this evaluation were KENO IV (2), NITAWL, and XSDRNPM (5). The reference microscopic cross section library was the 123 group super - XSDRN library, DLC-16 (6), which was obtained from the Radiation Shielding Information Center.

3.2 Cross Section Generation

The NITAWL and XSDRNPM Codes (5) were used to generate 16 neutron energy group cross sections. NITAWL was used to generate self shielded 123 group cross sections from the 123 group super-XSDRN library (DLC-16). The resulting working library is then collapsed into 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 and external moderators.

3.3 KENO IV Models

Homogenized fuel pin representation was utilized in the assembly interior. The cross shaped box, the outside moderator, tank wall, lattic grid, fuel pin lower plug, bottom plate and support plate were all explicitly represented. The structure details are shown in Figure 1 (which were obtained from Reference 1).

Table 2 summarizes the multiplication factors computed by KENO IV for 6 critical experiments together with the reported K_{eff} values as calculated in Reference 1 using the APPOLLO-MORET approach.

Among the five critical experiments which are analyzed only 3 had calculated APPOLLO-MORET results. F these three experiments the KENO IV, MORET APPOLLO and experi...ntal results agree within less than 0.5% K_{eff}.

3.4 Statistical Uncertainty and Bias

The statistical uncertainty and bias of the criticality analysis of the experiments have been calculated. The only criticality analysis 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 analysis. The results are as follows:

Total Number of Results	6
Mean Value (u)	1.00246
Standard Deviation	0.00432
2 Sigma	0.00864
Bias (u-1)	+0.00246
Uncertainty Minus Bias	0.00618

Based on the limted number of criticality experiments in the low hydrogen range, The calculational uncertainty to be applied to the criticality analysis will be increased to 0.03

TABLE 2

KENO IV Results for Noted Gap Widths

Description	Hydrogen Density gm/cm^3	K_{eff} KENO IV	APPOLLO-MORET
	Gap Width = 2.5 cm Between Assemblies		
Aluminum Box + Air	0.0	0.99641 ± 0.00407	1.005 ± .010
Aluminum Box + $(C_8H_8)n$	0.0025	0.99913 ± 0.00384	0.987 ± .010
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	1.006 ± .011
	Gap Width = 5.0 cm Between Assemblies		
Aluminum Box + Air	0.0	1.00412 ± 0.00422	------
Aluminum Box + Air	0.0020	1.00647 ± 0.00421	------
	Gap Width = 10.0 cm Between Assemblies		
Aluminum Box + Air	0.0	1.00117 ± 0.00390	0.996 ± 0.010
Aluminum Box	0.0022	1.00748 ± 0.00378	0.987 ± 0.010

Section 4

CONCLUSIONS

A 16 neutron energy group calculation model which employs a homogenized fuel assembly representation has been defined and verified against critical experiments. This model gives multiplication factors which are comparable to those obtained using the APPOLLO-MORET approach.

Section 5

REFERENCES

1. Dissolution and Storage Experiment with 4.75 wt% U(235) enriched UO_2 Rods, J. C. Manarandre et al, (Nuclear Technology Vol. 50, September 1980, Page 148.

2. L. M. Petrie and N. F. Cross, KENO IV, An Improved MONTE CARLO Criticality Program, ORNL-2938, November 1975.

3. W. R. Cable, 123-Group Neutron Cross Section Data Generated from ENDF/B-II Data for Use in the XSDRN Discrete Ordinates Spectral Averaging Code, DLB-16, Radiation Shielding Information Center, 1971.

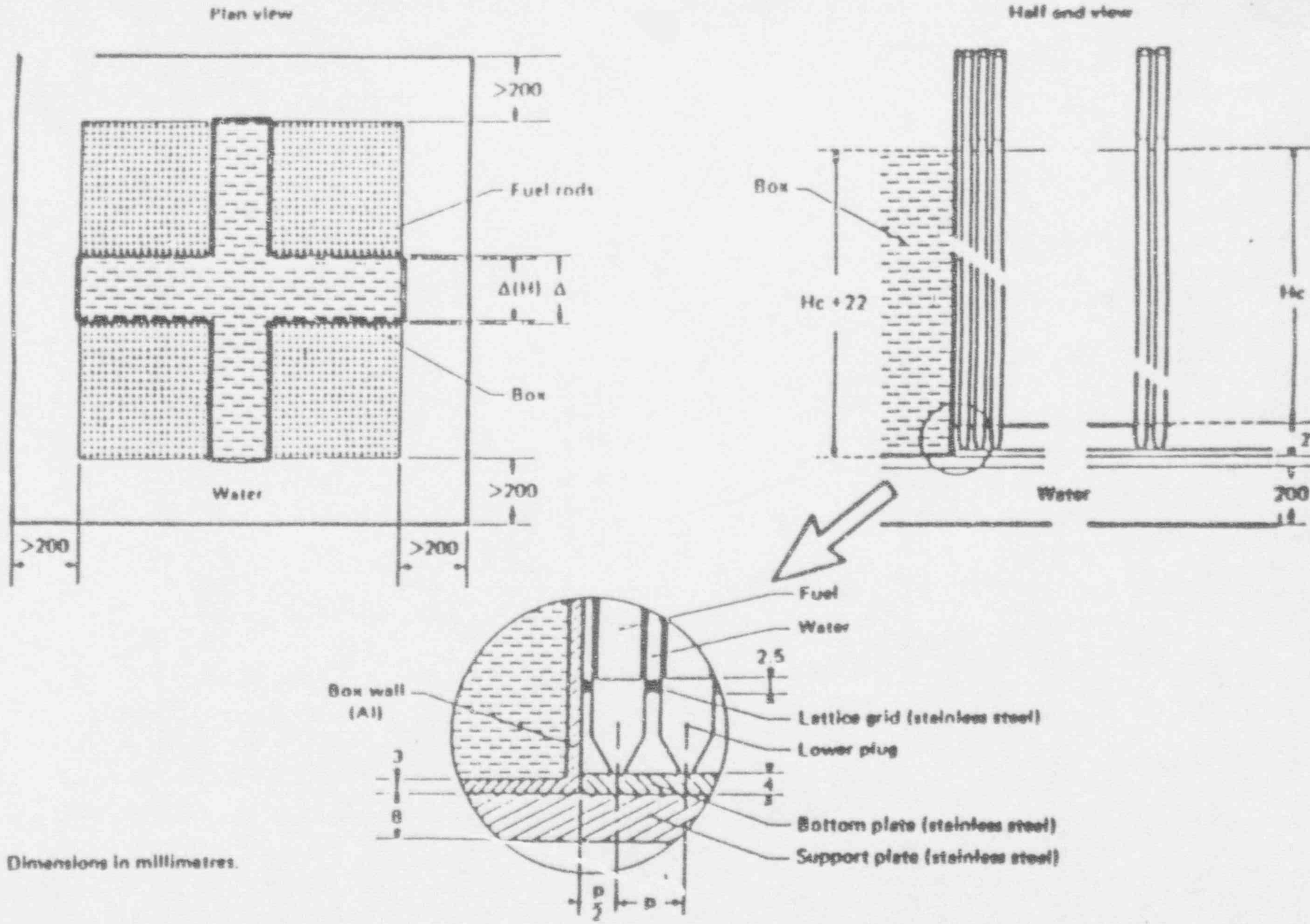
4. Scale: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation - Book II, NUREG/CR-0200.

5. N. M. Green et al, AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B, ORNL/TM-3706, March 1976.

6. 123-Group Neutron Cross Section in CCC-123/XSDRN Format Based on ENDF/B-II Data, DLB-16/COBB, Radiation Shielding Information Center. (These cross sections may be the same as in Reference 3, existing documentation received from RSIC does not allow a definitive conclusion).

7. A. M. Hathout et al, Validation of three Cross-Section Libraries Used With the Scale System for Criticality Safety Analysis, NUREG/CR-1917, June 1981.

Mana-andre et al, Dissolution and Storage Experimen



Dimensions in millimetres.

Figure 1

APPDENIX D

"MIST DENSITY CALCULATION FOR A SINGLE SPRINKLER HEAD"

Discharge from a single sprinkler head anywhere in region A, B, or C [see drawing NFM-C-4440," Layout of sprinkler system in Pellet Shop Annex (Region A) and Fuel Bundle Assembly Room (Regions B & C) in building 17].

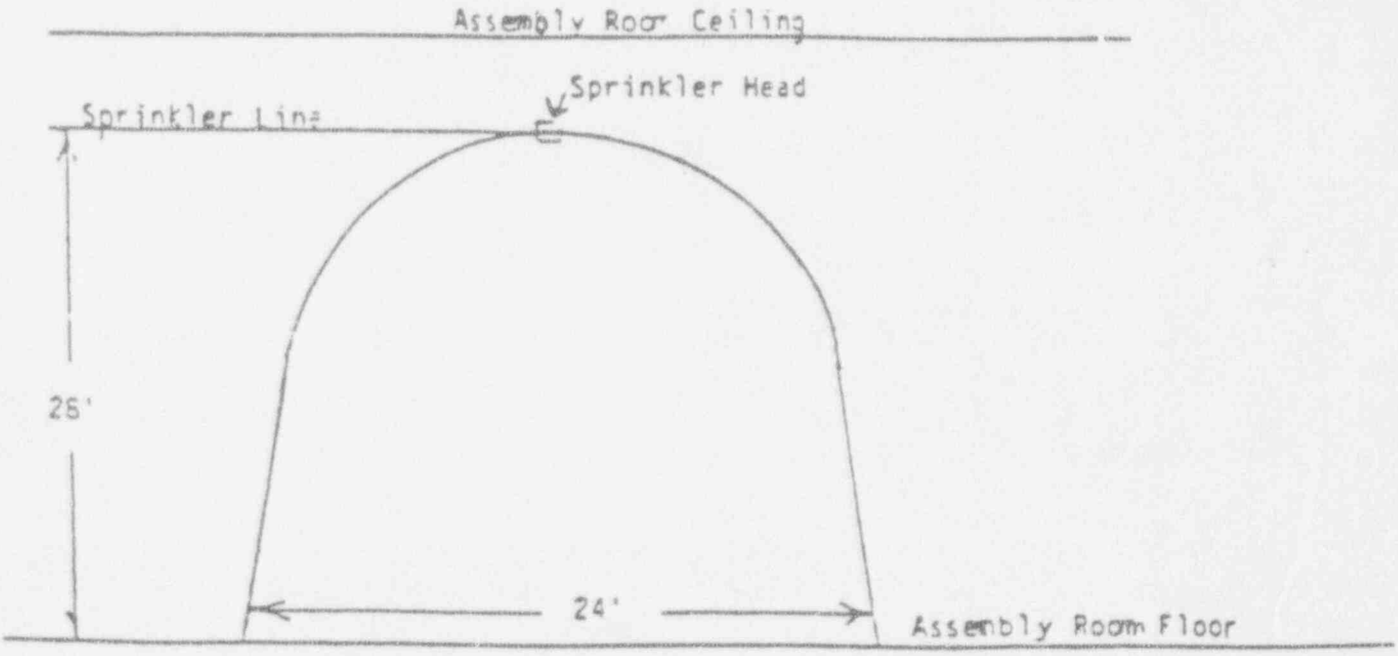
Discharge Flow from 1 Sprinkler = $Q = K(P)^{1/2}$

Where Q = Discharge Flow
K = Constant for 1/2" Sprinkler Head = 5.6
P = Discharge Pressure at head = 100 psi (assumes max. possible line pressure regardless of sprinkler location).

$Q = 5.6\ (100)^{1/2}$ = 56 gal/min (say 60 gal./min.)

= 60 gal/min. x 1 ft^3/7.5 gal.

Q = 8 ft^3/min.



 July 24, 1987

VOLUME OF SPRINKLER FLOW PARABOLOID = $V = 1/2\ \pi\ R^2 H$

Where π = A constant = 3.14

R = Radius of spray at floor = 12'

H = Distance from sprinkler head to floor = 28 ft.

$V = 1/2\ (3.14)\ (12)^2\ (28)$

$V = 6330\ ft^3$

WATER DROP SIZE FROM 1/2" SPRINKLER HEAD *

$$D_2 = \left(\frac{P_1}{P_2}\right)^{1/3} D_1$$

Where D_2 = Unknown water drop size

D_1 = Known water drop size = 0.86 mm

P_1 = Reference pressure @ known drop size = 30 psi

P_2 = Reference pressure @ unknown drop size = 100 psi

$$D_2 \text{ (Drop size @ 100 psi)} = \left(\frac{30}{100}\right)^{1/3} (0.86)$$

$$D_2 = (0.89)\ (0.86) = 0.77 \text{ mm @ 100 psi}$$

DROP VELOCITY

Reference drop velocity for 1 mm drop = 13 ft/sec *

Drop Velocity @ 0.77 mm = 0.77 x 13 = 10 ft/sec.

TIME FOR DROP TO REACH FLOOR

T = Drop time from 28ft - $\frac{28ft}{10ft}$ = 2.3 sec.

WATER VOLUME FRACTION

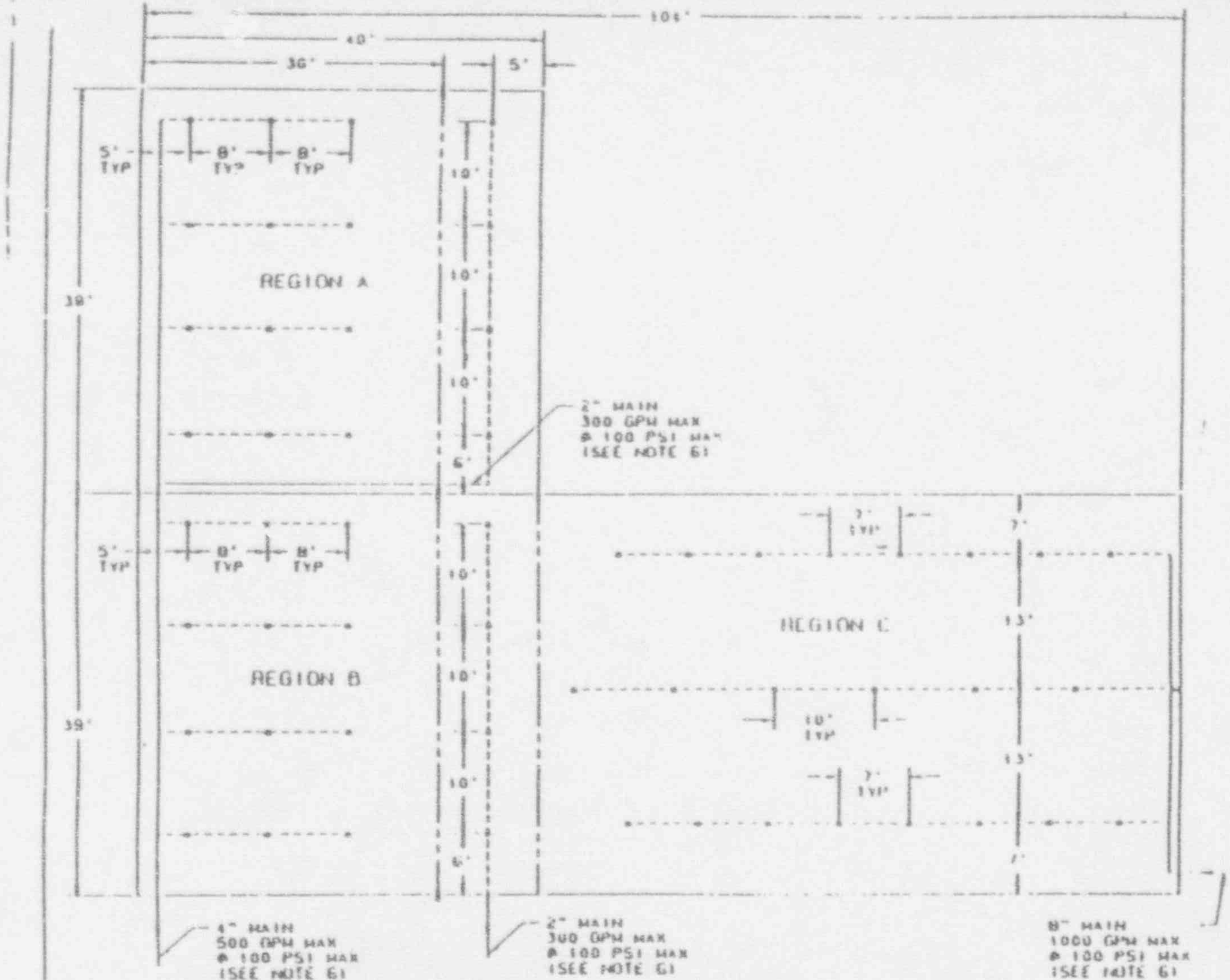
Water Volume Fraction = $\frac{QT}{V}$

Where Q = Discharge Sprinkler Flow in ft^3/min.
T = Time for Drop to go from Sprinkler to Floor
V = Paraboloid Volume

Water Volume Fraction = $\frac{(8)\ (2.8)\ (1/60)}{6330}$

Water Volume Fraction = 0.000059 Grams/cc

* Supplied by FMIC



6. SPRINKLER SYSTEM SUPPLIED BY 250,00 GALLON ELEVATED STORAGE TANK. PRESSURE ON SYSTEM IS MAINTAINED BY A MARATHON ELECTRIC PUMP (MODEL TD44STD02G) BACKED UP BY A [illegible] DIESEL PUMP (MODEL PTA-150-50) AND TWO PEERLESS PUMPS (MODEL 15A[illegible]

5. — · · — SIGNIFIES SEPARATION ZONE BETWEEN SPRINKLER LINES

4. - - - - SIGNIFIES SPRINKLER DISTRIBUTION LINE.

3. ——— SIGNIFI[illegible] MAIN WATER FEED LINE.

2. ALL SPRINKLERS ARE RELIABLE AUTOMATIC, MODEL G, 1/2" STANDARD ORIFICE, UPRIGHT UNIT.

1. UNLESS OTHERWISE SPECIFIED, ALL DIMENSIONS ARE IN FT.

NFM-C-4440

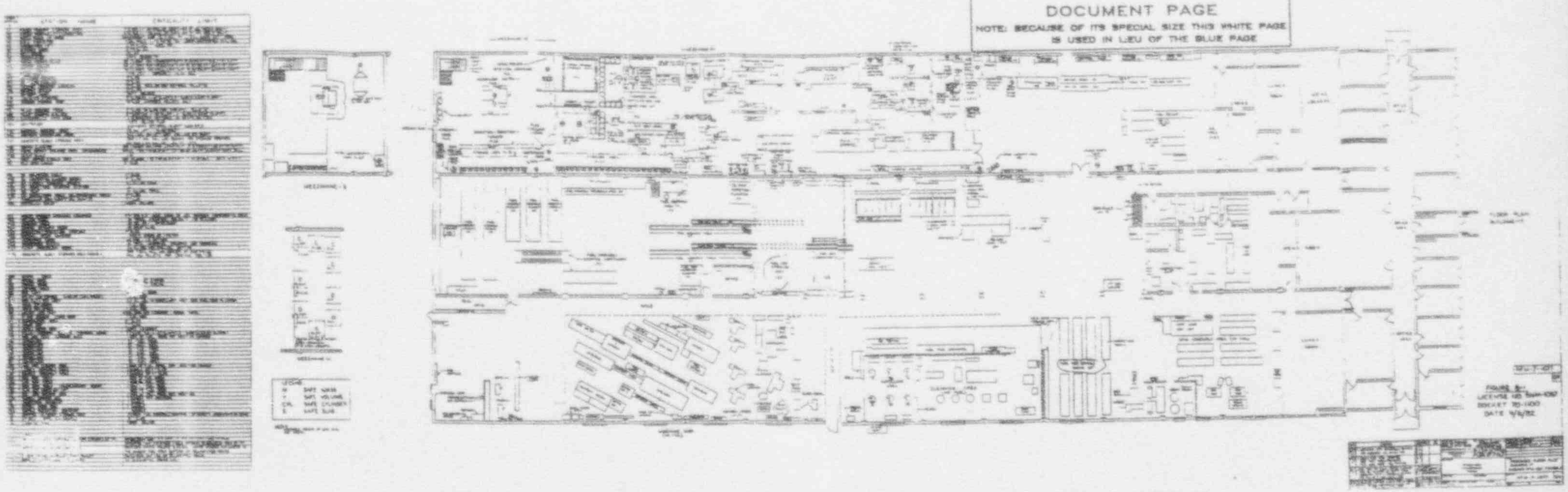
C-E POWER SYSTEMS	[illegible]	DRAWN [illegible]
CUSTOMER: NUCLEAR FUELS MANUFACTURING WINDSOR		TITLE: LAYOUT OF SPRINKLER SYSTEM IN PELLET SHOP AREA (REGION A) AND FUEL BUNDLE ASSEMBLY AREA (REGIONS B & C) IN BUILDING 17
NEXT ASSY	SUPERCEDED	DWG NO. NFM-C-4440 00
JOB NO. 776101	COMPONENT CODE	SCALE 1/8"=1'-0"

6

INTERNAL C-E NOTE:

OFFICIAL CONTROLLED DOCUMENT PAGE

NOTE: BECAUSE OF ITS SPECIAL SIZE THIS WHITE PAGE IS USED IN LIEU OF THE BLUE PAGE



ENCLOSURE III

WINDSOR PRODUCT DEVELOPMENT LABORATORY

NOTES:

1. The attached license application applies only to the product research and development activities (the "Laboratory License"). Certain activities, programs, and organizations are included that are common to or shared with the separated Manufacturing License SNM-1067.

2. This application is a complete document, which was administratively separated from SNM-1067 for the purpose of creating a separate Laboratory License. The attached is therefore submitted as Revision 0, with license identification to be determined by the NRC.

3. Suggested language covering the two license conditions requested: "Until September 30, 1993, the licensee shall limit the total enriched uranium possessed under this license to 3000 gms. of U-235. Effective September 30, 1993, this limit shall be reduced to 350 gms. U-235".