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CHAPTER 1 STANDARD CONDITIONS AND SPECIAL AUTHORIZATIONS

1.1 Name, Address and Corporate Information

The name of the applicant is Combustion Engineering, Inc. (C-E). The applicant is incorporated in the state of Delaware with its principal corporate offices located at 900 Long Ridge Road, Stamford, CT 06904. The address at which licensed activities will be conducted is:

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

1.2 Site Location

The site is located approximately 5 miles west of Windsor center on Prospect Hill Road which provides site access. Guard stations are provided at the points of vehicle entry to the portion of the site on which licensed activities are conducted. Fuel is fabricated inside a controlled access security area within the site. Section 1.5 identifies the buildings in which licensed activities will be conducted and summarizes the activities conducted in each.

1.3 License Number and Period of License

The number of the license to be renewed is SNM-1067, NRC Docket No. 70-1100. License renewal is requested for a period of ten (10) years.

1.4 Possession Limits

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

Isotope	Form	Quantity
1. Uranium enriched to ≤5.0 weight percent U235	Uranium Oxides	500,000 kg U
2. Uranium enriched to less than 20 weight percent U235	Any	4800 gms U235
3. Source material (Uranium)	Any	10,000 kgU

4. Pu238	Encapsulated Neutron Sources	5 sources, each containing less than 2.0 gm Pu238
5. Pu	Any Form as analytical samples	160 micrograms
6. Uranium enriched to ≥ 20 weight percent U235	Residue	4500 gms U235

1.5 <u>Authorized Activities</u>

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

Bldg. #3 & 3A - Storage and use of small quantities of radioactive material (<700 gms U235).

Bldg. 5 - Product development activities (≤740 gms U-235 enriched ≤5.0 weight percent, ≤350 gms U-235 enriched >5 weight percent.)

Bldg. 6 - Waste water processing from manufacturing and product development activities.

Bldg. 16 -Product development activities (<700 gms U235).

Bldg. 17 -Manufacture of fuel assemblies utilizing low enriched uranium (up to 5.0 weight percent U235).

Bldg. 18 -Product development activities (<700 gms U235).

Bldg. 21 -Storage of SNM in licensed shipping packages.

Windsor Site - Upon discovery of residues at the licensed facility containing uranium enriched to more than 19.99 weight percent U235, action shall be initiated within 30 days to remove the material from the premises in a timely manner. Such residues shall be stored in configurations demonstrated to be safe on the basis of criticality safety analyses performed in accordance with the requirements of Part I of this license application.

*Limits specified are for enriched material. They do not apply to depleted or natural uranium.

1.6 Exemptions and Special Authorizations

1.6.1 Transfer of SNM Between Buildings

Transfer of special nuclear material at the Windsor facility shall be handled in accordance with the approved Fundamental Nuclear Material Control Plan (FNMCP) referenced in Section 1.7 of this application.

1.6.2 Release of Materials & Equipment for Unrestricted Use (does not include the abandonment of buildings)

The release of equipment and material for unrestricted use shall be in accordance with "Guidelines for Decontamination of Facilities and Equipment Prior to Releases for Unrestricted Use or Termination of License for By-Product, Source, or Special Nuclear Material," USNRC, Annex B, August 1987. Annex B is provided in Appendix 3.1.

1.6.3 Leak Testing Sealed Sources Containing Alpha Emitters

Leak testing of encapsulated sources containing an alpha emitter shall be conducted in accordance with requirements derived from USNRC Annex A presented in Appendix 3.2.

1.7 Fundamental Nuclear Material Control Plan (FNMCP)

Combustion Engineering's FNMCP as revised by letters dated July 31 and August 7, 1989 was approved by the NRC as Safeguards Amendment SG-3 issued on October 16, 1989. The FNMCP, as modified by the above correspondence, is considered part of this license application.

CHAPTER 2 ORGANIZATION AND ADMINISTRATION

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

2.1 Organizational Responsibilities and Authority For Key Positions Important to Safety

2.1.1 Plant Manager

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

2.1.2 Director, Product Development

The Director, Product Development reports to the Vice President, Nuclear Fuel and is responsible for the management of nuclear fuel product development laboratory activities (SNM-1067). This responsibility encompasses the following functions: operations, accountability, security, training, criticality, radiological and industrial safety, environmental protection, materials handling and storage, and licensing.

2.1.3 Manager, Radiological Protection and Industrial Safety

The Manager, Radiological Protection and Industrial Safety reports to the Plant Manager. He is responsible for defining 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. He provides information, advice, and assistance to the operating and engineering Line Managers to ensure personnel and environmental protection measures are adequate. If the Manager believes any operation in the fuel manufacturing facility or product development laboratories to be unsafe, he has the authority to halt that operation. If an operation is halted for a safety reason(s) it shall not be restarted without his concurrence or that of the Plant Manager or a duly authorized alternate.

2.1.4 Nuclear Criticality Specialist

A Nuclear Criticality Specialist reports in a functional manner to the Manager, Radiological Protection and Industrial Safety. He advises Line Managers regarding criticality safety practices, arranges for analyses or reviews and approves changes to processes, procedures or equipment related to criticality safety. A Nuclear Criticality Specialist has the authority to halt any operation in the fuel manufacturing facility or product development laboratories that he believes to represent an unsafe criticality condition. If an operation is halted for a criticality safety reason(s) it shall not be restarted without his concurrence or that of the Manager, Radiological Protection and Industrial Safety or the Plant Manager or a duly authorized alternate.

2.1.5 Supervisor, Radiological Protection and Industrial Safety

The Supervisor, Radiological Protection and Industrial Safety reports to the Manager, Radiological Protection and Industrial Safety. He assists the Manager in carrying out his 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. If the Supervisor believes any operation in the fuel manufacturing facility or product development laboratories to be unsafe, he has the authority to halt that operation. If an operation is halted for a safety reason(s) it shall not be restarted without the concurrence of the Manager, Radiological Protection and Industrial Safety or the Plant Manager or a duly authorized alternate.

2.1.6 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.

2.1.7 Manager, Production

The Manager of Production reports to the Plant Manager. He is responsible for the planning, scheduling and control of the production process for the fabrication of fuel assemblies and their subsequent shipment to meet customer needs. Facility process/equipment operators are under the cognizance of the Manager of Production. He is responsible for ensuring the proper training of personnel and that procedures and safety limits are followed. The Manager of Production also oversees material and equipment purchasing, receiving, warehousing and inventory control.

2.1.8 Manager, Manufacturing Engineering

The Manager of Manufacturing Engineering reports to the Plant Manager. Engineering activities related to facility equipment, process, methods and construction, whether new or a modification are directed by the Manager of Manufacturing Engineering. The Manager is also responsible for equipment maintenance at the fuel manufacturing facility. He is also responsible for the preparation of procedures and training materials concerning facility equipment and the manufacturing process.

2.2 Personnel Education and Experience Requirements

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

2.3 Facility Review Group

The responsibility of the Facility Review Group is to periodically review nuclear fuel manufacturing and product development laboratories operational areas related to safety. The Group reports to the Plant Manager.

Items of consideration by the Group may include regulatory compliance, performance trends, program effectiveness, corrective actions, human factors, as well as other factors considered important by the Group or the Plant Manager.

The Plant Manager appoints the Group Chairman, and regular members may be selected by the Plant Manager or the Chairman. Alternates may also be appointed for each member. Each regular member and alternate will have at least five (5) years experience. The Group Chairman or Plant Manager may appoint temporary personnel and designate subcommittees as required to accomplish reviews. The Group Chairman or Plant Manager designates members who shall attend each meeting according to the topics to be considered.

The review frequency of operational areas shall be determined by the Facility Review Group Chairman. However, the minimum frequency shall be one (1) inspection per year for each of the following operational areas: radiological safety, criticality safety, ALARA, and internal inspection and audit related to safety.

The Facility Review Group shall issue a written report of its findings no later than one (1) month following each calendar quarter. The report shall contain results for all operational areas reviewed during the quarter. A minimum of one operational area will be reviewed each calendar quarter. Findings and recommendations (if any) of the Group shall be reported to the Plant Manager and Director of Product Development. Cognizant managers are responsible to correct, and document correction of, deficiencies identified by the Group. Reporting of Group findings shall be retained for a minimum of three (3) years.

2.4 Approval Authority for Personnel Selection

The Plant Manager and each of his direct reports, who are in key positions important to safety and are involved in activities within the scope of this application, 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.

The Director, Product Development and each of his direct reports, who are in key positions important to safety and are involved in activities within the scope of this application, 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.

2.5 Training

The degree of training an individual receives is commensurate with the potential conditions requiring radiological health protection to which he will be exposed. Escorted visitors do not require any training.

2.5.1 Initial Training

Employees and visitors requiring unescorted access to the Windsor Nuclear Fuel Manufacturing facility or to controlled radiological areas of the Product Development laboratories shall be indoctrinated in the safety aspects of the respective facilities. The indoctrination topics include nuclear criticality safety, radiation safety, industrial safety, ALARA practices and emergency procedures. After test results demonstrate that a new employee has sufficient knowledge in the above topics, the new employee begins job-specific training under direct line supervision and/or experienced personnel. The Supervisor monitors performance until it is adequate to permit work without close supervision.

Individuals who have had prior radiological protection training may be exempted from participation in the training program described above upon the successful completion of a challenge examination. Individuals not familiar with site specific information shall be instructed in those details to allow their safe conduct while at the facility.

General indoctrination training is provided to individuals in the NFM facilities who do not require entry into radiologically controlled areas. This training covers the following subjects:

1) NFM organization and administrative policies
2) Description of the manufacturing process and the quality control program and requirements
3) Security requirements
4) Industrial, radiation and criticality safety
5) Emergency procedures.

2.5.2 Retraining

Employees whose job involves working in radiologically controlled areas shall participate in a biennial, not to exceed twenty-five (25) months, Radiation Worker retraining program. The retraining program shall emphasize the key safety aspects of their jobs and shall include, as a minimum, criticality safety and radiological safety.

2.5.3 Training Records

Formal training sessions shall be documented and competency demonstrated by passing a test to verify training effectiveness. 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 and Product Development laboratory operations which directly involve licensed materials shall be conducted in accordance with written procedures. The preparation, review, revision, approval and implementation of safety related operating procedures shall be accomplished through a document control system. The minimum frequency for review, for the purpose of updating, of operating procedures involving Special Nuclear Materials shall be every two (2) years. Updating of operating procedures is the responsibility of the Cognizant Manager.

2.7 Audits and Inspections

Audits and inspections shall be performed to determine if plant operations involving licensed materials are conducted in accordance with regulations, applicable license conditions, and written procedures. Audits and inspections shall apply to safety-related and environmental programs. Personnel having no direct responsibility for the specific plant operation being audited or inspected shall be used.

The inspection function is a normal part of the Radiological Protection and Industrial Safety Technician job. As such, the inspection function is informally satisfied on regular basis. Additionally, Technicians shall perform a documented inspection monthly using a prepared checklist to review the conduct of facility operations. Most problems are normally corrected on the spot by cognizant shift personnel. More significant problems are reported to the cognizant manager as deficiencies on an inspection report, along with the corrective action taken or to be taken. The cognizant manager is responsible for corrective action and its documentation.

Quarterly inspections or audits, performed by the RP&IS Manager or his designated representative, cover criticality control, radiation safety and industrial safety. The review of criticality control shall be performed by an individual having at least the requirements of a Nuclear Criticality Specialist. Items requiring corrective action are documented in a report distributed to the Plant Manager and manager level staff. The Cognizant Manager is responsible for corrective action, except where another manager is specifically designated. Follow-up actions taken by the responsible manager shall be documented.

An annual audit is conducted in which the results of previous inspections or audits are reviewed, as an evaluation of the effectiveness of the program. The audit is documented by a formal report to the Vice President, Nuclear Fuel. The annual audit is performed by a team appointed by the Vice President, Nuclear Fuel. The team shall include, as a minimum, a Nuclear Criticality Specialist and a radiation specialist who shall audit criticality and radiation safety. The annual audit will review ALARA requirements in conformance with Regulatory Guide 8.10, as applicable. The Cognizant Manager shall be responsible for follow-up of recommendations made by the audit team and documentation of corrective action.

2.8 Investigations and Reporting

Events specified by applicable regulations or license conditions are investigated in accordance with written procedures and are reported to the Plant Manager or the Director, Product Development, as appropriate.

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. Evaluation, corrective action and documentation of a reportable event are the responsibility of the cognizant manager.

2.9 Records

Retention of records required to be maintained by the regulations, and by the conditions of this license, shall be the responsibility of the cognizant manager. A procedure indicates the types of records retained and retention time.

TABLE 2-1 Minimum Education And Experience Requirements For Key Personnel

Described In Section No.	Position Title	Education	Experience (Years/Field)
2.1.1	Plant Manager	Bachelors, Science or Engineering	10 Total, 5/Nuclear industry management
2.1.2	Director, Product Development	Bachelors, Science or Engineering	10 Total, 5/Nuclear industry management
2.1.3	Manager, Radiological Protection and Industrial Safety	Bachelors, Science or Engineering	5/Health Physics, with 2/Operational health physics with uranium bioassay techniques, internal exposure control, and radiation measurement techniques
2.1.4	Nuclear Criticality Specialist	Bachelors, Science or Engineering	2/Nuclear criticality evaluations
2.1.5	Supervisor, Radiological Protection and Industrial Safety	High School Diploma	5 Total/Nuclear industry, with 3/Senior Health Physics Technician
2.1.6	Radiological Protection and Industrial Safety Tech. (non-entry level)	High School Diploma	2/Training and experience in Radiation Protection activities
2.1.7	Manager, Production	High School Diploma	10 Total/Nuclear industry, with 5/nuclear fuel manufacturing including 3/Production Coordination
2.1.8	Manager, Manufacturing Engineering	Bachelors, Engineering	3/Engineering design of process, systems or facilities

CHAPTER 3 RADIATION PROTECTION

3.1 Special Administrative Requirements

3.1.1 ALARA Commitment

It is the policy of Combustion Engineering to maintain an on-going program to ensure the safety, health and well-being of its employees. In implementing this policy, Combustion Engineering supports the philosophy of maintaining radiation exposures to the general public, its employees and the environment "as low as reasonably achievable", ALARA.

For activities carried out within the scope of this application, responsibility for establishing and ensuring adherence to this policy shall rest with the Plant Manager and the Director, Product Development. This policy shall be implemented through appropriate delegations to the Manager, Radiological and Industrial Safety, the applicable Line Managers, and the Facility Review Group.

3.1.2 Radiation Work Permit Procedures

All work with radioactive materials shall be performed under the control of a Radiation Work Permit (RWP) or written procedures. In accordance with an established administrative procedure, safety-related procedures involving work with radioactive materials shall be approved by the Radiological Protection Supervisor or the Manager, Radiological Protection and Industrial Safety (RPIS). Depending on the complexity and routine nature of the work to be performed, specific task procedures may be invoked by an RWP. The RWP is initiated by the Cognizant Engineer or supervisor and approved by the Manager or Supervisor RPIS. The RWP describes the work to be performed and specifies the necessary safety requirements.

RWP's shall be reviewed for their need every 30 days as a minimum. The Manager or Supervisor of Radiological Protection and Industrial Safety shall close out each RWP upon completion of the work for which it was issued.

3.1.3 Written Procedures

Activities related to radiation protection shall be governed by written procedures. Procedures shall be made available to appropriate personnel through a document control system.

3.2 Technical Requirements

3.2.1 Restricted Areas-Personnel Contamination Control

3.2.1.1 Radiological Control Areas

Radiological control areas designating radiation areas, airborne radioactivity areas, contamination control areas and radioactive material areas are established as needed. Signs are posted to appropriately identify the areas. Radiological conditions for each controlled area are stated in written procedures or by RWP.

3.2.1.2 Change Rooms

Change rooms are provided at the access points to the portion of Building 17 in which unclad fuel material is present. A visible boundary is provided between the radiologically controlled area of the change rooms and the adjacent uncontrolled areas of the facility.

3.2.1.3 Protective Clothing

Protective clothing is required for activities in which personnel may become contaminated. The clothing required depends on the activity being performed and on the kind and amount of radioactive material present. Protective clothing requirements are specified in written procedures or RWP's.

3.2.1.4 Personnel Monitoring Systems

Alpha radiation detectors are provided at the exit from contaminated areas. Personnel must survey their personal clothing and exposed body surfaces upon exiting a contaminated area.

3.2.1.5 Personnel Decontamination Policy

Personnel must wash their hands upon leaving a contaminated area. If the required survey of clothing and exposed body surfaces produces an instrument reading above background, the individual shall promptly notify a member of the Radiological Protection and Industrial Safety staff and not leave the area until they respond and the situation is resolved.

3.2.2 Ventilation

Nuclear Fuel Manufacturing

The ventilation systems have been designed to maintain a negative pressure differential between the portion of Building 17 in which unclad fuel material exists and other areas. The direction of air flow shall be checked monthly and documented.

In addition, to assure that 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.

Whenever a pressure drop of 4 inches of water is measured across the combination of the prefilter and first bank of HEPA filters, action shall be taken to reduce the pressure drop to < 4 inches of water. Each ventilation system is provided with instrumentation that continuously measures the pressure drop. The pressure drop shall be checked monthly and documented. When the face velocity at a ventilated hood drops below 100 fpm, action will be taken 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 monthly.

Following all filter changes or other movement of filters in the fixed air systems, a Technician from the Radiological Protection and Industrial Safety staff shall inspect the placement of the filters for proper sealing. In addition, air samples will be taken and counted immediately after approximately 1/2, 2, and 8 hours of operation to assure the filters are adequately filtering the exhaust air.

Product Development

Stacks used for the exhausting of radioactive effluents from Product Development (Building 5) are equipped with sampling stations and HEPA filters. The exhaust is continuously monitored whenever operations involving dusting or release of radioactive material are in progress.

If airborne activity results, averaged over a two week period, exceed 25% of the applicable concentration listed in Table II, Column 1 of 10 CFR 20 Appendix B for air being discharged to an unrestricted area, an investigation will be conducted and corrective actions taken.

Whenever a pressure drop of $\geq$ 4 inches of water is measured across the combination of the prefilter and first bank of HEPA filters, action shall be taken to reduce the pressure drop to < 4 inches of water.

Each ventilation system is provided with instrumentation that continuously measures the pressure drop. The pressure drop for all systems shall be checked monthly and documented. When the face velocity at a ventilated hood drops below 100 fpm, action will be taken to increase the air flow to 100 fpm minimum or the hood will not be used to handle radioactive material. Face velocities will be checked monthly. After all filter changes or movement of filters, the filters shall be tested either by 1) counting samples immediately after approximately 1/2 hour of operation or 2) DOP testing the filters in accordance with ANSI standards. The results of these tests shall be documented.

3.2.3 Work Area 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 when work is performed in the area. Air sampling shall be accomplished using fixed position air sampling stations located strategically throughout the shop and/or lapel air samplers. Lapel air samplers shall be used, if necessary, by individuals who work with or handle uncontained licensed material. If lapel air samplers are used for these workers, then lapel air sample results will be used for the basic evaluation of the workers' internal exposure. In lieu of lapel air samplers, the results from the fixed position breathing zone air samplers may be used for the basic evaluation of the internal exposure of individuals working with uncontained licensed material if the representativeness of the fixed position samplers has been validated. The representativeness of the fixed position air sampling stations shall be validated at least once every 12 months after initial validation. The average results from the fixed position air samplers shall be used for the basic evaluation of all individuals except those using lapel air samplers. All samples from lapel and fixed position air sampling shall be analyzed after each working shift.

During the normal operating period, if a lapel air sampler or a fixed position air sampling station indicates the airborne concentration of radioactivity for that work area exceeds the MPC as specified in Table I Column 1 of 10 CFR 20, Appendix B, an investigation as to the cause shall be conducted. Any necessary corrective actions to prevent recurrence shall be taken and documented.

Whenever an individual's seven (7) consecutive day assigned internal exposure exceeds 32 MPC hours, he shall be closely monitored to preclude exceeding 520 MPC hours in a calendar quarter.

Product Development

When the monitoring of airborne concentration of radioactivity is required as specified in 10 CFR 20.103, the air concentration of radioactivity in Product Development shall be analyzed within 24 hours after each operating shift. The required monitoring shall be conducted by breathing zone samples 100% of the time. If a sample indicates that the airborne concentration of radioactivity in a work area exceeds the MPC as specified in Table I Column I of 10 CFR 20 Appendix B, an investigation to determine the cause shall be conducted.

3.2.4 Radioactivity Measurement Instruments

3.2.4.1 Counting and Survey Instruments

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 the Radiological Protection and Industrial Safety program. Additional instrumentation is maintained for emergency use as outlined in the Emergency Plan referenced in Part I, Section 8. Instruments are calibrated semi-annually and following any repair that affects the accuracy of the measurements. The calibration of the survey instruments shall be performed in accordance with the manufacturer's recommended procedure. Daily source response checks and battery checks are performed on survey instruments in use. The alpha counting equipment is checked daily to verify background and efficiency.

3.2.4.2 Criticality Alarm System

A criticality alarm system which meets the requirements of 10 CFR 70.24 (a) (1) and the guidance of Regulatory Guide 8.12, "Criticality Accident alarm System" shall be maintained in Building 5 Product Development areas and the Building 17/21/6 areas. The radiation intensity is shown on a panel located in a non-radiologically controlled area in Building 17 for Buildings 6, 17 and 21, and on a panel in a non-radiologically controlled area in Building 5 of Product Development. There is an alarm which serves as a radiation evacuation alarm. The monitors are connected to the emergency power system.

Operational tests of the radiation monitors are performed monthly by Radiation Protection and Industrial Safety personnel. The alarm setpoint shall be checked annually and following any repair that affects its setpoint accuracy.

3.2.5 Radiation Exposures

Programs for determining, validating and controlling occupational exposures will be conducted.

To monitor external exposure, each individual who enters a radiologically controlled area under circumstances such that he is likely to receive a dose in any calendar quarter in excess of 25 percent of the applicable value specified in 10 CFR 20.101 (a) shall be supplied with a TLD badge for purposes of personnel dosimetry. Badges will be processed quarterly. 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 and neutron foils are also strategically placed throughout the facility to record background radiation levels and radiation resulting from a criticality accident. These TLD badges will also be processed quarterly during normal operations and immediately following a criticality accident.

Further measures to monitor and control radiation exposures are described in Section 3.2.3, Work Area Air Sampling, Section 3.2.6, Surface Contamination and Section 3.2.7, Bioassay Program.

3.2.6 Surface Contamination

3.2.6.1 Radiologically Controlled Areas

The action levels and actions to be taken for removable alpha contamination are:

Alpha Level	Action
$\geq$ 10,000 dpm/100 cm.^{2}	Immediate clean-up
$\geq$ 5,000 dpm/100 cm^2 but < 10,000 dpm/100 cm^2	Delayed clean-up

Fixed contamination shall be limited as required to control external radiation exposures.

Surveys shall be conducted weekly using smears and survey meters.

3.2.6.2 Non-radiologically Controlled Areas

The action levels and actions to be taken for removable alpha contamination are:

Alpha Level	Action
$\geq$ 200 dpm/100 cm.^{2}	Immediate clean-up
$\geq$ 100 dpm/100 cm.^{2} but < 200 dpm/100 cm^2	Delayed clean-up
$\geq$ 10 dpm/100 cm.^{2} (lunch rooms only)	Immediate clean-up

Fixed alpha contamination shall be less than 500 dpm/100 cm.^{2} average.

Surveys shall be conducted monthly except for the lunch rooms which shall be surveyed daily. Smears and survey meters are used.

3.2.6.3 Release of Materials and Equipment

Decontamination of facilities and equipment prior to release for unrestricted use or termination of licenses for special nuclear material shall be performed in accordance with USNRC Annex B dated August, 1987. A verbation copy of Annex B is provided in Appendix 3.1.

3.2.6.4 Leak Testing Seal Sources Containing Alpha Emitters

Leak testing of encapsulated sources containing an alpha emitter shall be conducted in accordance with requirements derived from USNRC Annex A presented in Appendix 3.2.

3.2.7 Bioassay Program

The bioassay program is used to verify the effectiveness of the internal exposure control program and to evaluate the potential uptake of an individual under accident or abnormal conditions. The minimum routine bioassay program consists of annual urinalysis and in-vivo measurements. Urinalysis will be performed monthly for workers who routinely work with Class D compounds of uranium.

APPENDIX 3.1

USNRC Annex B, August, 1987

PART I
LICENSE CONDITIONS

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

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

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

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

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

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

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

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

a. Identify the premises.
b. Show that reasonable effort has been made to eliminate residual contamination.
c. Describe the scope of the survey and general procedures followed.
d. State the findings of the survey in units specified in the instruction.

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

Table 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

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

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

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

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

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

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

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

APPENDIX 3.2

Requirements Derived from USNRC Annex A

Leak testing of encapsulated sources containing an alpha emitter shall be conducted in accordance with the following requirements derived from USNRC Annex A.

Each sealed source containing licensed material shall be tested for leakage at intervals not to exceed six months. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, a sealed source received from another person shall not be put into use until tested.

The periodic leak test required above does not apply to sealed sources that are stored and not being used. The sources shall be tested for leakage prior to any use or transfer to another person unless they have been leak tested within six months prior to the date of use or transfer.

The test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently mounted or stored on which one might expect contamination to accumulate. Records of leak test results shall be kept and maintained for inspection by the NRC.

If the test reveals the presence of 0.005 microcurie or more of removable contamination, the sealed source shall be immediately withdrawn from use and shall be decontaminated and repaired by a person appropriately licensed to make such repairs or be disposed of in accordance with NRC regulations.

Within five days after determining that any source has leaked, a report shall be filed with the Division of Industrial and Medical Nuclear Safety, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, describing the source, the test results, the extent of contamination, the apparent or suspected cause of source failure, and the corrective action taken. A copy of the report shall be sent to the Administrator of the nearest NRC Regional Office having jurisdiction listed in Appendix D of Title 10, Code of Federal Regulations, Part 20.

CHAPTER 4 NUCLEAR CRITICALITY SAFETY

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

Administrative conditions define:

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

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

c) the written procedures and postings employed to define the approved processes for handling and storage of SNM.

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

4.1 Administrative Conditions

4.1.1 Process Design Philosophy

The process design philosophy employed by Combustion Engineering, Inc. to assure nuclear criticality safety is based on the following key elements.

a) Process design, in so far as the handling and storage of SNM, shall incorporate sufficient factors of safety that at least two unlikely, independent, and concurrent changes in process conditions are required before a criticality accident may occur.

b) Physical controls and permanently engineered safeguards shall be the preferred method of criticality control so as to reduce dependence on administrative procedures. In some processes, types of control other than safe geometry, e.g., moderation, concentration, and/or poison, may be employed to achieve adequate process throughput. In these cases, all controlled parameters and their limits shall be clearly specified, approved by management in their review and approval of postings and operating procedures, and communicated to all affected personnel through postings, operating procedures, and training.

c) Before a new operation with SNM is begun or an existing operation is changed, it shall be determined that the entire process will be subcritical under normal and credible abnormal conditions.

4.1.2 Positions Responsible for Criticality Safety

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

All proposed changes or modifications to SNM processing, handling, or storage equipment or related operations in Nuclear Manufacturing shall be approved for criticality safety. Manufacturing Engineering shall process all change requests for Nuclear Manufacturing and secure the necessary management and safety approvals and reviews prior to implementation of the change. Facility change requests affecting criticality areas in Product Development shall be initiated by and processed by a designee of the Director, Product Development. These change requests shall have management and safety approvals prior to implementation of the change. If a nuclear criticality safety evaluation is required, it shall be carried out by a Nuclear Criticality Specialist. He may request the support of Nuclear Criticality Analysts in the Fuel Engineering Department.

4.1.3 Documenting Criticality Evaluations and Reviews

Criticality evaluations associated with facility changes affecting the handling and storage of SNM in Nuclear Manufacturing and the criticality areas in Product Development shall be documented and reviewed.

The criticality evaluations shall specify all criticality limits and controls; these limits and controls shall be incorporated into applicable written procedures and postings. Production and line supervisory personnel shall assist in the preparation of written procedures and postings. Day-to-day monitoring of workers for conformance to criticality limits and controls and administrative procedures is carried out by line supervisors and Radiological Protection and Industrial Safety personnel.

Documentation shall be sufficiently detailed and unambiguously presented that an independent reviewer can reconstruct the analysis and bases for the conditions presented. These conditions shall include all assumptions affecting criticality safety process limits and controls. If explicit analyses using validated computer methodology are employed, the margin to criticality along with a clear definition of assumed off-nominal conditions shall be provided.

All criticality evaluations shall be reviewed by a qualified independent reviewer, i.e., an individual who qualifies as a criticality specialist and did not participate either in the specification of the mode of analysis or the actual analysis of the facility change. The independent review shall be documented.

Records of the criticality evaluation and the independent review shall be maintained according to the requirements of applicable internal procedures as well as Section 2.9 of this License.

Product Development activities in Buildings 3, 3A, 16, and 18 are not included under the provisions of this section. The more stringent authorized limits on the amount of SNM for Buildings 3, 3A, 16, and 18 (700 grams U-235 per building) preclude criticality safety concerns.

4.1.4 Written Procedures

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

a) Traveler - This document specifies a sequence of operations required to process a given material, component, or assembly.

b) Operation Sheets - An Operation Sheet specifies the requirements of how a given step, operation, or process must be performed. It specifies required process parameters and methods. It is specified by number in a Traveler when it is required.

c) Radiation Work Permits - The radiation work permit is employed for those jobs involving the handling and/or storage of SNM which are not covered by standard procedures or which may involve increased potential for radiological exposure. A radiation work permit may supplant operating procedures in a development and testing environment.

4.1.5 Posting of Limits and Controls

All work stations and storage areas shall be posted with the nuclear safety limits and controls applicable to that station or area and approved by the Manager of Radiological Protection and Industrial Safety and a Nuclear Criticality Specialist. The latter Manager shall maintain a current record of: 1) the review and approval of each posting, 2) the location of each posting, and 3) the content of each posting.

Production and line Supervisors shall monitor the day-to-day conformance of individual workers to the posted limits and controls.

4.1.6 Labeling of Special Nuclear Material

All containers employed in the handling or storage of special nuclear material shall be labeled as to their contents. If SNM material is in the container, the amount, enrichment and form (powder, pellets, clean or dirty scrap, etc.) shall be specified; if empty, the container shall be so labeled or placed in designated areas for empty containers.

The labeling of containers shall be consistent with approved operating and/or test procedures for the purposes of providing visible conformance to nuclear criticality controls and limits applicable to specific process lines, storage areas, etc.

4.1.7 Preoperational Testing and Inspection

Prior to startup of a new or modified process in Nuclear Manufacturing and subsequent to the criticality safety evaluation and preparation of written procedures and postings, an inspection of equipment, procedures, and postings shall be carried out by representatives of Manufacturing Engineering and the Manager of Radiological Protection and Industrial Safety or the Nuclear Criticality Specialist to assure completeness and consistency between safety evaluations, equipment design, written procedures, and postings. This inspection shall be documented as part of the records for this facility change.

4.1.8 Criticality Safety Design

Internal procedures, reviewed and approved by management, assure that all new processes or changes in existing processes affecting the handling and storage of special nuclear material are evaluated for nuclear criticality safety. In the Nuclear Fuel Manufacturing Facilities these procedures require all facility changes affecting the handling and storage of SNM be executed through Manufacturing Engineering with appropriate criticality safety reviews.

In Product Development, criticality safety is assured through controls on facility changes affecting the handling and storage of SNM in specified criticality areas, the movement of SNM, restrictions on the amount of SNM in any designated criticality area to a conservative safe mass of U-235, and a minimum spacing of twelve feet between each defined criticality area.

4.2 Technical Criteria

4.2.1 Individual Units

4.2.1.1 Safe Individual Units (SIU)

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

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

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

4.2.1.2 Subcritical Units

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

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

4.2.1.3 Criteria

a) The possibility of accumulation of fissile materials in not readily accessible locations shall be considered. The potential for accumulation shall be either eliminated through equipment design or included in the nuclear safety evaluation of the process.

   Periodic cleanup shall serve as a basis for assessing changes in equipment design, operating procedures, or bases for nuclear safety evaluations should any accumulation be identified.

b) Nuclear safety evaluations shall include all credible sources of internal moderation, including process variables.

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

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

e) Reactivity hold-down by other than fixed poisons shall not be employed in criticality evaluations. Use of enhanced structural parasitics, e.g., boron stainless steel, shall be contingent upon a program to periodically verify the presence of the parasitic additive.

f) All storage racks, furnaces, containment, and processing equipment which provide nuclear safety limiting parameters shall be designed to assure against failure under normal and reasonable overload conditions and under conditions of shock or collision foreseeable in the plant area. 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 to fire and corrosion.

4.2.2 Multiple Units and Arrays

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

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

4.2.2.1 Spacing of Safe Individual Units

The following criteria shall be employed:

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

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

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

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

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

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

f) Vessels and other items of equipment requiring exclusion areas shall have the limits of these areas clearly marked on the floor should they extend beyond the boundaries of the equipment. SNM in transit shall not be permitted to enter an exclusion area unless the SNM is entering that work station. This rule shall be covered in operator training and operating procedures.

4.2.3 Technical Data

4.2.3.1 Safe Individual Unit Limits

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

Spacing criteria are given in Table 4-2 for the SIU limits of Table 4-1. Mass limited units may be stacked on a vertical centerline with at least a 10-inch separation. Maximum allowed volume for stacked units shall be 20 liters.

Table 4-3 summarizes applicable safe limits based on concepts noted in Section 4.2.1.2, i.e. criteria other than minimum critical values and fraction critical.

4.2.3.2 Other Criteria

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

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

c) A 35 Kg mass limit may be employed for homogeneous or heterogeneous UO_2 in a covered, five gallon, or less, stainless steel container. Heterogeneous UO_2 shall include hard, clean scrap, i.e., broken pellets and chips; hard, dirty scrap shall be limited to the SIU values listed in Table 4-1. These containers shall be separated by a minimum of twelve inches, edge to edge, in a planar array; alternate separation distances may be defined by analysis as in the case of interposed media.

4.2.4 Special Controls

The following technical criteria shall be employed.

a) UO_2 pellet thickness on each of the Pellet Storage Shelves shall meet the slab limit specified in Table 4-1. The shelves shall be covered from the top to prevent flooding of the pellet containers by overhead water sprays.

b) Fuel rod storage boxes are limited to an overall fuel rod stack height of 6 inches. A single fuel rod storage box meeting the 6 inch stack height shall be treated as a SIU. Two or more fuel rod storage boxes meeting the stack height limit shall meet the requirements of Table 4-3. Fuel rod storage boxes in the positioning table of the fuel rod pre-stacking station shall meet a close packed, hex pitch requirement.

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

d) An overhead sprinkler system as well as portable extinguishers are located throughout the Fuel Manufacturing Facilities and Product Development. Onsite and offsite fire protection service personnel have been instructed to use only portable dry chemical extinguishers in Bldg. #17 to maintain the highest possible margin of nuclear criticality safety. Fire hoses shall not be permitted in Bldg. #17.

e) All storage containers containing UO2 outside of hoods and in transit or in storage spaces shall be covered. Any storage containers accidentally internally moderated shall be handled as individual mass units and shall be stored in the concrete block storage area pending disposition.

f) Incoming pellets in the UNC-2901 shipping containers shall be stored either within Building 21, the transport vehicle or the Building 21/17 security complex. If stored within the vehicle, said vehicle shall be within the security complex. Two UNC-2901 containers are strapped to a pallet in a horizontal position. Three pallets can be stored within the Building 17 Pellet Shop Annex and four can be stored in the Building 17 Pellet Loading Area. Only one UNC-2901 shipping container on each shipping pallet shall be opened at a time for unloading. The contents of the open shipping container must be emptied before opening the other shipping container on the pallet. The exclusion area assigned to a pallet with an open shipping container shall be at least 25 ft^2. Fissile material, other than the contents of the shipping containers on the pallet being unloaded, shall be excluded from the 25 ft^2 area.

g) The amount of UO2 contained in the following hoods shall be limited as indicated below:

General Purpose Hood - $\leq$ 35.0 Kgs UO2 in right side
$\leq$ 35.0 Kgs UO2 in left side
Mass units separated by at least 1 foot.

Filter Knockdown Hood - $\leq$ 35.0 Kgs UO2 in upper portion
$\leq$ 35.0 Kgs UO2 in lower portion
Mass units separated by at least 1 foot.

h) 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".

i) The fuel assemblies in the storage positions of the fuel assembly storage room, when wrapped with polyethylene, shall have the bottom ends open to assure drainage. Fire fighting in the fuel assembly storage room with fire hoses is prohibited.

j) Loaded new fuel assembly shipping containers may be stored outdoors in arrays up to three high. Containers shall be stored on pavement or blacktop within a 8 foot high chain link fence.

k) Waste containers shall be stored in designated areas of the pellet shop, on a concrete pad continguous 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 U-235 each; arrays will meet the surface density criteria.

l) Designated criticality areas in Building 5 shall be limited to either $\leq$ 740 grams U-235 for U-235 enrichments $\leq$ 5 w/o or $\leq$ 350 grams U-235 for enrichments > 5 w/o. Criticality areas shall be separated by a minimum of 12 feet. A continuous log shall be maintained for each criticality area in Product Development to assure the limit is maintained and that the enrichment of all material is recorded.

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

(1) 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 g H_2O/cc) as demonstrated in Section 14.7.
(2) Criteria used in the choice of fire protection in areas of potential criticality accidents shall be justified in writing.

(3) An audit of the existing fire sprinkler system in Building 17 shall be conducted once a quarter (Sprinkler Heads, Risers, Distribution Lines, and Pumps) to see to it that it has not been modified or added to in any way that would impair its performance or have an effect on calculated mist density.

(4) All proposed changes to the fire sprinkler system, that could affect Building 17 will be reviewed and approved in accordance with the procedures described in Section 2.6 as regards facility changes affecting criticality for their effect on mist density, before such changes are implemented.

(5) Plastic bags which are placed around the fuel assembly shall be left open at the bottom at all times.

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

(7) In any area where unsealed hard clean scrap or randomly loaded pellet containers are exposed to the fire sprinkler system they will be assumed to fill with water. Hard dirty scrap shall be assumed as optimally moderated.

(8) Possible moderating material around fissile materials will be included in the analysis.

TABLE 4-1

SIU LIMITS MEETING FRACTION CRITICAL CRITERIA

U-235 Enrichment w/o		Mass Limits (KgUO2) Homogeneous	Heterogeneous
>Nat.	2.5	54	50
>2.5	3.0	41	38
>3.0	3.2	36	36
>3.2	3.4	35	33
>3.4	3.6	32	30
>3.6	3.8	28	27
>3.8	4.1	24	24
>4.1	4.3	22	22
>4.3	4.5	20	20
>4.5	4.7	18	18
>4.7	5.0	16	16
		Volume (L)	
>Nat.	3.5	31	22
>3.5	4.1	25	18
>4.1	5.0	22	17
		Cyl. Dia. (in.)	
>Nat.	3.5	10.7	9.5
>3.5	4.1	9.8	8.9
>4.1	5.0	9.2	8.4

		Slab Thickness (in) Homogeneous	Heterogeneous Corrugated Trays	Heterogeneous Randomly Loaded Boats
>Nat.	3.5	4.0	4.4	4.0
>3.5	4.1	4.0	3.9	4.0
>4.1	4.3	4.0	3.7	4.0
>4.3	5.0	4.0	3.5	4.0

TABLE 4-2

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

	Homogeneous		Heterogeneous	
	Limit	SIU Spacing Area (ft2)	Limit	SIU Spacing Area (ft2)
Mass (1)	6.2 $KgUO_2/ft^2$	2.6	5.8 $KgUO_2/ft^2$	2.8
Volume (2)	1.1 in.	8.5	1.0 in.	7.2
Cylinder (2)(3)	1.1 in.	5.1	1.0 in.	4.6

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

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

(3) Per foot of cylinder height.

TABLE 4-3 Subcritical Process (Storage) Limits Not Meeting Fraction Critical Criteria

Location	Limit Type	Max. Value	Restrictions
Bldg. 17	Mass-Pellets, Clean Hard Scrap	35 Kg	Closed, $\leq$ 5 gal. S.S. Container
	Fuel Rod Stack Height in Fuel Rod Storage Boxes	6.0 inches	None on pitch except rod pre-stacking station (3).
	Mass > 5 w/o U-235	350 g U-235	
	UNC-2901 Container Pallets	7(2)	One Container per pellet open at a time, one foot minimum spacing from process equipment.
Bldg. 21	UNC-2901 Array Size	3 pallets high	None on surface area
	Mass	350 g U-235	> 5 w/o U-235
Bldg. 17/21 Security Area	UNC-2901/927 Al/Cl Mixed Array Size	3 units high	None on array size
Bldg. 5	Mass per designated Criticality Area(1)	740 g U-235,	$\leq$ 5 w/o U-235
		350 g U-235	> 5 w/o U-235
	Slab-Stacked Rods	3.5 inches	
Bldg. 6	Mass per tank	740 g U-235	
Bldg. 3,3A,16,18	Mass per building	700 g U-235	$\leq$ 5 w/o U-235

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(1) Criticality areas separated by a minimum of 12 feet.
(2) 4 in pellet loading area, 3 in pellet shop annex.
(3) Fuel rod storage boxes in positioning table shall meet a close packed, hex pitch spacing of contained fuel rods.

CHAPTER 5 ENVIRONMENTAL PROTECTION

5.1 Effluent Control Systems

5.1.1 Airborne Effluents

Control equipment, action levels and ALARA provisions for airborne radioactive effluents are described in Section 3.2.2, Ventilation.

In the event that the calculated dose to any member of the public in any consecutive 12-month period is about to exceed the limits specified in 40 CFR 190.10, Combustion Engineering shall take immediate steps to reduce emissions so as to comply with 40 CFR 190.10. As provided in 40 CFR 190.11, a petition to the Nuclear Regulatory Commission may be made for a variance from the requirements of 40 CFR 190.10. If a petition for a variance is anticipated, the request shall be submitted at least 90 days prior to exceeding the limits specified in 40 CFR 190.11.

5.1.2 Liquid Effluents

Effluent control systems shall be used to maintain releases of radioactive material in liquid effluents to unrestricted areas below the limits specified in 10CFR20.106. Administrative limits below those of 10CFR20.106 shall be established. Procedures governing liquid effluent discharge ensure that regulatory limits are met.

5.2 Environmental Monitoring Program

5.2.1 Fallout Stations

Stations for collecting rainfall and particulate fallout are distributed on the Windsor site property. These samples shall be collected and analyzed quarterly 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 industrial stream and points upstream and downstream from the confluence of the industrial stream and the Farmington River. These samples shall be collected and analyzed quarterly 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 collected and analyzed quarterly for gross alpha and gross beta radioactivity and total uranium.

5.2.4 Vegetation and Soil Samples

Vegetation and soil samples shall be collected and analyzed semi-annually at each of the fallout station locations on-site and four location in the grassy area surrounding Building #17. Additional samples are collected off-site in locations generally to the north, south, east and west of the site. These samples shall be analyzed for gross alpha and gross beta radioactivity and total uranium.

CHAPTER 6 SPECIAL PROCESSES

6.1 Proprietary Information

No proprietary information is included in this license application.

6.2 Occupational Safety

Combustion Engineering follows the current American Conference of Governmental Industrial Hygienists, Occupational Safety and Health Administration and Nuclear Regulatory Commission maximum permissible concentrations, threshold value limits and permissible exposure limits for radioactive and hazardous chemicals in the design and operation of its fuel fabrication facility.

In case of a known release, Radiological Protection and Industrial Safety (RPIS) personnel are contacted to ascertain the concentration levels and the recommended personnel protective equipment required for cleanup operations to proceed. RPIS personnel conduct routine or periodic surveys as appropriate to determine the concentrations of routinely utilized radioactive and hazardous chemicals.

6.3 Emergency Utilities

6.3.1 Emergency Electrical Power Supply

An emergency generator provides back-up power for emergency lighting and fire and radiation alarms.

6.4 Radioactive Waste Management

6.4.1 Airborne Radioactive Effluents

Control of airborne radioactive effluents is described in Section 3.2.2, Ventilation.

6.4.2 Liquid Radioactive Effluents

Control of liquid radioactive effluents is described in Section 5.1.2, Liquid Effluents.

6.4.3 Solid Radioactive Waste

Wastes contaminated with radioactive material to levels in excess of limits specified in USNRC ANNEX B (see Appendix 3.1) shall be packaged and transferred to a licensed waste processor and/or licensed disposal facility in accordance with applicable requirements specified in Title 10 and Title 49 of the Code of Federal Regulations.

Waste packages are stored on an outside pad within the fenced area of the Building 17/21 complex. No single package will contain more than 350 grams of U235. Packages will not be opened outdoors. The waste packages will contain no liquid wastes.

The waste packages will be secured against release of contamination and will be provided with a tamper-indicating seal. The packages will be labeled as to enrichment and U235 content.

The storage area will be surveyed quarterly for contamination and the packages will be inspected to verify their integrity.

CHAPTER 7 DECOMMISSIONING

Combustion Engineering reaffirms that, upon terminating activities involving materials authorized under license SNM-1067, affected facilities will be decommissioned in a manner that will protect the health and safety of the public. Requirements for financial assurance for decommissioning and for decommissioning plans are specified in 10 CFR 70.25 and 10 CFR 70.38, respectively.

CHAPTER 8 RADIOLOGICAL CONTINGENCY PLAN

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

CHAPTER 9 GENERAL INFORMATION

9.1 Corporate Information

The names, positions, addresses and citizenship of Combustion Engineering's principal officers are shown in Table 9-1.

On December 15, 1989, Asea Brown Boveri Inc. (ABB) announced that it had accepted for payment approximately 96.7 percent of the outstanding shares of common stock of Combustion Engineering, Inc. ("the applicant"). It is anticipated that the merger of Combustion Engineering with ABB will be approved at a forthcoming meeting of Combustion Engineering stock holders.

The fuel fabrication organization, including the relationship and responsibilities of key positions important to safety, is described in Chapter 2. The organization structure is depicted in Figure 11-1. The President, Nuclear Power Businesses shown on Figure 11-1 reports to the Vice President, Power Services Businesses identified as a principal officer in Table 9-1.

9.2 Financial Qualification

Combustion Engineering's 1988 10-K report which details its financial position is provided at the end of the chapter as Appendix 9.1. Provisions for decommissioning are described in Chapter 7.

9.3 Summary of Intended Post-Redeployment Operating Objective and Process

9.3.1 Operating Objective

The operating objectives of activities covered by this license application are:

1. To produce nuclear reactor fuel enriched to $\leq$ 5% U235, and
2. To conduct product development activities involving the use of special nuclear material.

9.3.2 Process Description

9.3.2.1 Fuel Manufacturing Process

Pellets enriched to ≤ 5% U235, are received in approved containers and are either placed in storage or sent to a drier as operational needs dictate. Following drying, the pellets are either stored or sent directly to the stacking tables as required.

The stacking tables contain troughs into which pellets are loaded until the required pellet stack length is attained. The pellet stacks are then pushed into rods on which the lower end cap has previously been welded. The upper end caps are welded on the loaded rods, and the rods are then leak tested, inspected to verify conformance to various physical parameters and stored.

Rods are withdrawn from storage as required and inserted into the fuel assembly grids during assembly fabrication. Completed assemblies are then stored in Building 17 or placed in approved shipping containers for storage outside or shipment off-site.

9.3.2.2 Product Development

With respect to activities covered by this license, Product Development maintains the facilities and staff necessary to use special nuclear material in design, development, analysis and testing in support of various Combustion Engineering products and services.

9.3.3 Expected Post-Redeployment Process Changes Since Last License Renewal

Pellet production operations have been deleted from the fuel fabrication process in Building 17. This has resulted in removal of powder storage facilities, powder preparation equipment, pellet presses, dewaxing and sintering furnaces, pellet grinders and roller micrometers. Scrap recycle has also been deleted from the fuel fabrication process. Clean and dirty scrap is shipped to Hematite for recycling. The fuel fabrication process is described in detail in Chapter 15.

9.4 Site Description

9.4.1 Location

The site is located in the Town of Windsor, County of Hartford in the State of Connecticut. It is approximately 5 miles west of the center of the Town of Windsor and approximately 9 miles north of the City of Hartford. Its location within Connecticut and in relation to surrounding communities is shown in Figure 9-1.

9.4.2 Topography

Elevations in the vicinity of the site vary from about 100 feet above sea level at the Farmington River to hilltop heights up to 289 feet, with the level areas being at an elevation averaging approximately 210 feet above sea level. The land rises to the west of the site to a north-south ridge approximately two miles distant which averages 450 feet in elevation. The land slopes gently away from the site on all sides as evidenced by the direction of flow of small streams which drain the area.

9.4.3 Demography

The area surrounding the site is sparsely populated. Windsor, Connecticut, within which the site is located, has a population of 26,420 and a population density of 892 per square mile. East Granby, Connecticut is approximately three miles away with a population of 4,280 and a population density of 240 per square mile. The distribution of population in the area is shown in Table 9-2. Figure 9-1 is a map of the general area showing the location of the towns listed in Table 9-2.

9.4.4 Meteorology and Climatology

Meteorological data show that the mean temperature for the Hartford area is approximately 50.5 F. The minimum and maximum monthly mean temperatures are 18.4 F and 83.6 F, respectively. The mean annual precipitation is about 42 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 Engineering site is approximately 180 feet above mean sea level, the probability of direct damage from a local flood is very low.

The highest recorded wind velocity was 66 miles per hour in September, 1985. The prevailing wind direction for the six months from May through October is South, and for the six months from November through April is Northwest. The average wind velocity at the 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 conditions occur with low air flow.

The area's location relative to the continent and the ocean influences its meteorological and climatological conditions. With the prevailing west-to-east air flow, continental modifications of the air are important. However, sudden 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". Seasonal air mass characteristics vary from the extremely cold and dry continental polar quality of winter to the warm, humid, maritime tropical characteristics of summer.

Local topography also influences the weather. The Berkshire Hills to the west and northwest are a source of summer thunderstorms which can be accompanied by wind and hail. Frequently during the winter, icing results when rain falls through the cold air trapped in the Connecticut Valley. On clear nights in the late summer or early autumn, cool air drainage into the Valley, plus Connecticut River moisture, sometimes produce ground fog.

9.4.5 Hydrology

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

The site creek, into which site effluents are discharged, flows into the Farmington River which flows along the northwest corner of the site. 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.

9.4.6 Geology/Seismology

The surrounding area has been subjected to the actions of glacial ice. Dominant geological features are a result of erosion and depositions caused during the Pleistocene era.

The State of Connecticut has a favorable earthquake history. Ten earthquakes are recorded, the first in 1791 and the last in 1925. All of these, with the exception of the first, were local in nature and of moderate intensity. The Town of Windsor is in earthquake zone 2.

9.5 Location of Buildings on Site

The location of buildings on the site is shown on Figure 9-2. The buildings in which licensed activities are performed are: 3, 3A, 5, 6, 16, 17, 18 and 21. Buildings 5, 6, 17, and 21 are described below because these are the buildings in which licensed activities are principally conducted and in which the bulk of special nuclear material is located. Only small quantities (<700 gms. U-235) are used or stored in each of the other buildings.

9.5.1 Building 5 Description

Building 5, the Nuclear Product Development Facility, contains 60,000 square feet of floor space. The building is arranged as a main bay with three wings extending off the bay at right angles. The wings are designated the north, central and south wings. The building is sided principally with corrugated asbestos panels and has a poured gypsum roof of various heights. A high bay structure 100 feet in height is attached to the central wing at the end opposite the main bay. Although the high bay is formally designated as Building 18, it is physically and operationally integral with Building 5 and its area is included in the 60,000 square feet noted above.

The main bay and each of the three wings contain office space occupying a total of 27,000 square feet.

Mechanical testing and research/development facilities occupy a total of 16,500 square feet of space in various portions of the main bay, the central wing including the high bay and the south wing.

Facilities for an electronics testing laboratory, fabrication of research test fuel and chemical testing of production fuel for quality control purposes occupy an additional 16,500 square feet in the main bay and three wings.

9.5.2 Building 6 Description

Building 6, the Hot Waste Retention Vault, is a poured concrete structure housing tanks for receiving radioactive liquid wastes and diluting them prior to discharge. The building contains 10 retention tanks of 2,000 gallons capacity each and 4 dilution tanks, each having a capacity of 5,000 gallons.

9.5.3 Building 17 Description

Building 17, the Fuel Fabrication Facility, measures 120 feet by 340 feet (40,800 square feet). The shop section of the building measures 120 feet by 300 feet (36,000 square feet), has a concrete floor, corrugated asbestos siding and a poured gypsum roof approximately 25 feet above floor level. The office section in the front of the building is 120 feet by 40 feet (4,800 square feet), has a concrete floor, concrete block exterior walls with full height windows and a poured gypsum roof deck approximately 11 feet above floor level. Trailers providing additional office space are located in front of Building 17 and are connected to the building by an enclosed corridor.

9.5.4. Building 21 Description

Building 21 is a warehouse used to store incoming fuel shipping containers, raw materials used in the fuel fabrication process and finished components. The building is a prefabricated steel frame structure approximately 120 feet by 80 feet. The sides and roof are corrugated metal and the floor is concrete.

9.6 Maps and Plot Plans

Figure 9-3 shows the location within the site of buildings in which activities covered by this license application are conducted. The security area around Buildings 17 and 21 is also shown. Figure 9-4 shows the immediate vicinity of the site and Figure 9-2 shows the buildings on the site.

Prominent natural features in the site vicinity are described in Section 9.4.2, Topography. Man-made features of significance are: Bradley International Airport approximately 4 miles to the north; the Griffin Office Center, a 10 building complex adjacent to the site on the west; the Knolls Laboratory, adjacent to the site on the east.

The distance and direction to surrounding population centers are shown in Table 9-2.

9.7 License History

License number SNM-1067 was originally issued on May 29, 1968 and was renewed on January 30, 1975 and March 14, 1983.

Table 9-1 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, Jeffery S.	Vice President of Finance	900 Long Ridge Road Stamford, CT	U.S.
SMITH, Dale E.	Vice President Human Resources & Operations	900 Long Ridge Road Stamford, CT	U.S.
CONNOLLY, William J.	Vice President Corporate & Investor Relations	900 Long Ridge Road Stamford, CT	U.S.
KISTNER, Robert E.	Vice President Information Systems and Services	900 Long Ridge Road Stamford, CT	U.S.
MASON, Robert H.	Vice President Venture Finance & International	900 Long Ridge Road Stamford, CT	U.S.
SANDERSON, Jack T.	Vice president Corp. Technology	900 Long Ridge Road Stamford, CT	U.S.
GARRY, Bernard J.	Secretary	900 Long Ridge Road Stamford, CT	U.S.
JONES, Fred R.	Treasurer	900 Long Ridge Road Stamford, CT	U.S.
INSLEY, Preston E.	Controller	900 Long Ridge Road Stamford, CT	U.S.

Table 9-2 Population and Population Density of Area Towns

Town	Approximate Distance of Town Center to Site	General Direction from Site	Square Miles	Estimated Population (1985)	Population Density Person/sq mile
Hartford	9	S	18.6	135,200	7,268
Windsor	5	SE	29.6	26,420	892
Bloomfield	5	S	26.9	19,810	736
West Hartford	8	SW	21.6	60,790	2,838
East Hartford	9	SE	18.2	52,320	2,874
Manchester	13	S	27.6	50,660	1,835
South Windsor	7	SE	29.2	19,790	677
East Windsor	6	SE	25.6	9,100	342
Windsor Locks	4	E	9.6	12,260	1,277
East Granby	3	N	17.8	4,280	240
Simsbury	6	SW	34.2	22,750	665
1/3 Avon incl. Center	8	SW	7.4	4,170	553
Granby Center	6	N	10.0	8,600	860
1/2 Suffield incl Center & W. Suffield	6	N	22.0	5,090	236
1/2 Ellington excl Center	12	E	17.0	5,145	294
1/3 Vernon excl Center	13	E	6.0	9,650	1,564
Totals			333.3	460,445	

APPENDIX 9.1

FORM 10-K ANNUAL REPORT

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

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

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

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

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

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

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

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

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

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

Yes X No____

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

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

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

Documents Incorporated By Reference:

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

Exhibit index located at sequential page number ________.

PART I

ITEM 1. DESCRIPTION OF BUSINESS

(a) General Development of the Business

References to the Company or C-E contained herein refer to Combustion Engineering, Inc. and/or any one or more of its subsidiaries. References to the Annual Report refer to the Annual Report to Shareholders of Combustion Engineering, Inc. for the year ended December 31, 1988.

The Company is a worldwide supplier principally to the power and process industries and to municipalities and government agencies. Its principal markets include: electric utilities, independent power projects, pulp and paper, petrochemical, chemical and pharmaceutical, food and beverage, primary metals, minerals and mining, and textiles. Its major products and services include: (a) for the power industry - power plant systems and services, operating software, emission control and heat recovery equipment and construction and project management; (b) for process industries - measurement and control systems; instrumentation; engineering, operating and management services; petrochemical and refinery technology; pulping equipment; specialty minerals and refractories; and screening, grinding and milling equipment; and (c) for the public and environmental sectors - municipal waste-to-energy systems, hazardous waste site cleanup, environmental consulting assessment and monitoring and hazardous waste systems; operating maintenance and training services; and mass transit engineering and construction services.

The Company's commitment is to be the best value producer worldwide for power generation, process industries and the public sector and environmental - offering products and services that optimize quality, cost and usefulness. To best follow through on this mission and remain responsive to changing market conditions, the Company has continued to focus globally as well as to restructure its operations and reposition its businesses.

International sales in 1988, 1987 and 1986 accounted for 40, 33 and 30 percent of the Company's total revenues. Given the increasing global nature of its business opportunities, C-E has taken a number of steps to improve its ability to compete effectively in global markets. In January 1989, C-E and Alsthom S.A. of Paris announced a proposal to form a joint venture combining both companies' businesses supplying fossil-fueled steam supply systems and related parts and maintenance services to electric utilities and industrial users worldwide. This alliance should prove particularly advantageous in positioning for the increasing golbalization of the power generation market. In addition, C-E has made progress with its initiatives with the Soviet Union and has won a number of projects in such countries as the People's Republic of China, Korea, Thailand and Japan.

In restructuring itself, C-E has made a number of acquisitions and divestitures over the past few years. In November 1988, the Company acquired the assets of Energy Systems Group (ESG) from systems Control Inc.

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

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

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

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

(b) <u>Financial Information about Industry Segments</u>

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

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

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

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

(c) <u>Narrative Description of Business</u>

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

Raw Materials

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

Patents and Licenses

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

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

Backlog

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

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

The Power Generation and Public Sector and Environmental segments' backlogs, however, decreased from their record levels in 1987 as work was carried out in 1988 on 1987 orders (including the nuclear plants in Korea, circulating fluid bed plants in the United States, other fossil plants in the United States, Canada and the People's Republic of China, and waste-to-energy plants in the United States). The Power Generation segment made significant contributions to its backlog with an award for a 150 megawatt circulating fluid bed unit in the United States and for two 300 megawatt lignite-fired steam generators in northeastern Thailand, valued at $82,000,000.

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

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

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

Competitive Conditions

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

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

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

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

Research and Development

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

Compliance with Environmental Protection Laws

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

Employees

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

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

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

ITEM 2. **PROPERTIES**

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

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

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

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

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

ITEM 3. PENDING LEGAL PROCEEDINGS

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

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

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

None.

OFFICERS OF THE COMPANY

Listed below are the officers of the Company:

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

There are no family relationships among the foregoing officers.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

PART II

ITEMS 5. THROUGH 8.

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

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

ITEM 9. DISAGREEMENTS ON ACCOUNTING AND FINANCIAL DISCLOSURE

None.

PART III

ITEM 10. DIRECTORS AND EXECUTIVE OFFICERS OF THE COMPANY

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

ITEM 11. EXECUTIVE COMPENSATION

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

ITEM 12. SECURITY OWNERSHIP OF CERTAIN BENEFICIAL OWNERS AND MANAGEMENT

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

ITEM 13. CERTAIN RELATIONSHIPS AND RELATED TRANSACTIONS

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

PART IV

ITEM 14(a). EXHIBITS AND FINANCIAL STATEMENT SCHEDULES

Documents: Page

NOTES:

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

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

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

Form 8-K dated December 30, 1988

Item 5. Other Events
Adoption of Shareholder Rights Plan.

SIGNATURES

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

COMBUSTION ENGINEERING, INC.

By ______________________
Charles L. Hugel
Chairman and Chief Executive Officer, Director

By ______________________
Jeffrey S. Rubin
Vice President - Finance

By ______________________
Preston E. Insley
Controller

Dated March 29, 1989

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

Lucy Wilson Benson *
By ______________________
Director

Paul W. MacAvoy *
By ______________________
Director

Walter H. Helmerich, III *
By ______________________
Director

Scott L. Probasco, Jr. *
By ______________________
Director

Robert M. Jenney *
By ______________________
Director

Robert C. Seamans, Jr. *
By ______________________
Director

By Arthur J. Santry Jr
Arthur J. Santry, Jr.
Director

Robert G. Stone, Jr. *
By ______________________
Director

George S. Kimmel*
By ______________________
President and Chief Operating Officer, Director

Kenneth J. Whalen *
By ______________________
Director

David R. Whitwam *
By ______________________
Director

*Pursuant to Power of Attorney

Dated: March 29, 1989

By Bernard J. Garry
Bernard J. Garry,
as Attorney-In-Fact

Exhibits

		Sequential Page No.
(3) Articles of incorporation and by-laws		
(3)-1	Restated Certificate of Incorporation of Combustion Engineering, Inc.	Incorporated by reference to form 10-Q Report for the second quarter 1987 and form 8-K filed December 30,1988
(3)-2	By-Laws of Combustion Engineering, Inc., as amended June 27, 1988	Incorporated by reference to Form 10-Q Report for second quarter 1988
(10) Material contracts		
(10)-1	Acquisition agreement among Hughes Tool Company, Combustion Engineering, Inc., Combustion Engineering Limited, C-E Europe Limited and Vetco Gray Inc., together with certain exhibits and amendments	Incorporated by reference to Form 8-K Report dated November 14, 1986
(10)-2	Agreements incident to the acquisition of AccuRay Corporation	Incorporated by reference to Form 8-K Report dated January 29, 1987
(10)-3	Amended Incentive Compensation Plan, as amended July 24, 1986	Incorporated by reference to Form 10-Q Report for the third quarter of 1986
(10)-4	Deferred Compensation Plan for Non-Employee Directors of Combustion Engineering, Inc., as amended November 19, 1986	Incorporated by reference to Form 10-K Report for 1986
	Amendment adopted December 19, 1988	
(10)-5	Deferred Compensation Plan, as amended November 19, 1986	Incorporated by reference to Form 10-K Report for 1986
	Amendment adopted December 19, 1988	
(10)-6	Executive Retirement and Life Insurance Plan, as amended July 24, 1986, and Form of Life Insurance Agreement	Incorporated by reference to Form 10-K Report for 1985 and to Form 10-Q Report for the third quarter of 1986
	Amendment adopted December 19, 1988	

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

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

(13) Annual Report to Shareholders for the Year Ended December 31, 1988

(22) Subsidiaries of the Registrant

(24) Consent of Experts

(25) Powers of Attorney

(25)-1	Power of Attorney for Lucy W. Benson and David R. Whitwam	Incorporated by reference to Form 10-K Report for 1987
(25)-2	All Other Powers of Attorney	Incorporated by reference to Form 10-K Report for 1985 and 1983

ARTHUR ANDERSEN & CO.
STAMFORD, CONNECTICUT

REPORT OF INDEPENDENT PUBLIC ACCOUNTANTS ON SCHEDULES

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

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

Arthur Andersen & Co.
ARTHUR ANDERSEN & CO.

Stamford, Connecticut,
February 17, 1989

SCHEDULE I

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

YEAR ENDED DECEMBER 31, 1988
(Dollars in thousands)

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

				$164,099
				========
Other Investments				
At Cost				$ 26,099
At Equity				48,450

				$ 74,549
				========

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

SCHEDULE II

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

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

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

Schedule VII

COMBUSTION E[illegible]INEERING, INC. AND SUBSIDIARY COMPANIES
GUAR[illegible]S OF SECURITIES OF OTHER ISSUERS

[illegible]CE[illegible]ER 31, 1988

([illegible] thousands)

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

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

SCHEDULE IX

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

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

(1) Includes drawdowns utilized to purchase AccuRay.

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES

SUBSIDIARIES OF THE REGISTRANT

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

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

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

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

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

ARTHUR ANDERSEN & CO.

STAMFORD, CONNECTICUT

EXHIBIT 24

CONSENT OF INDEPENDENT PUBLIC ACCOUNTANTS

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

Arthur Andersen & Co.

ARTHUR ANDERSEN & CO.

Stamford, Connecticut,
March 29, 1989

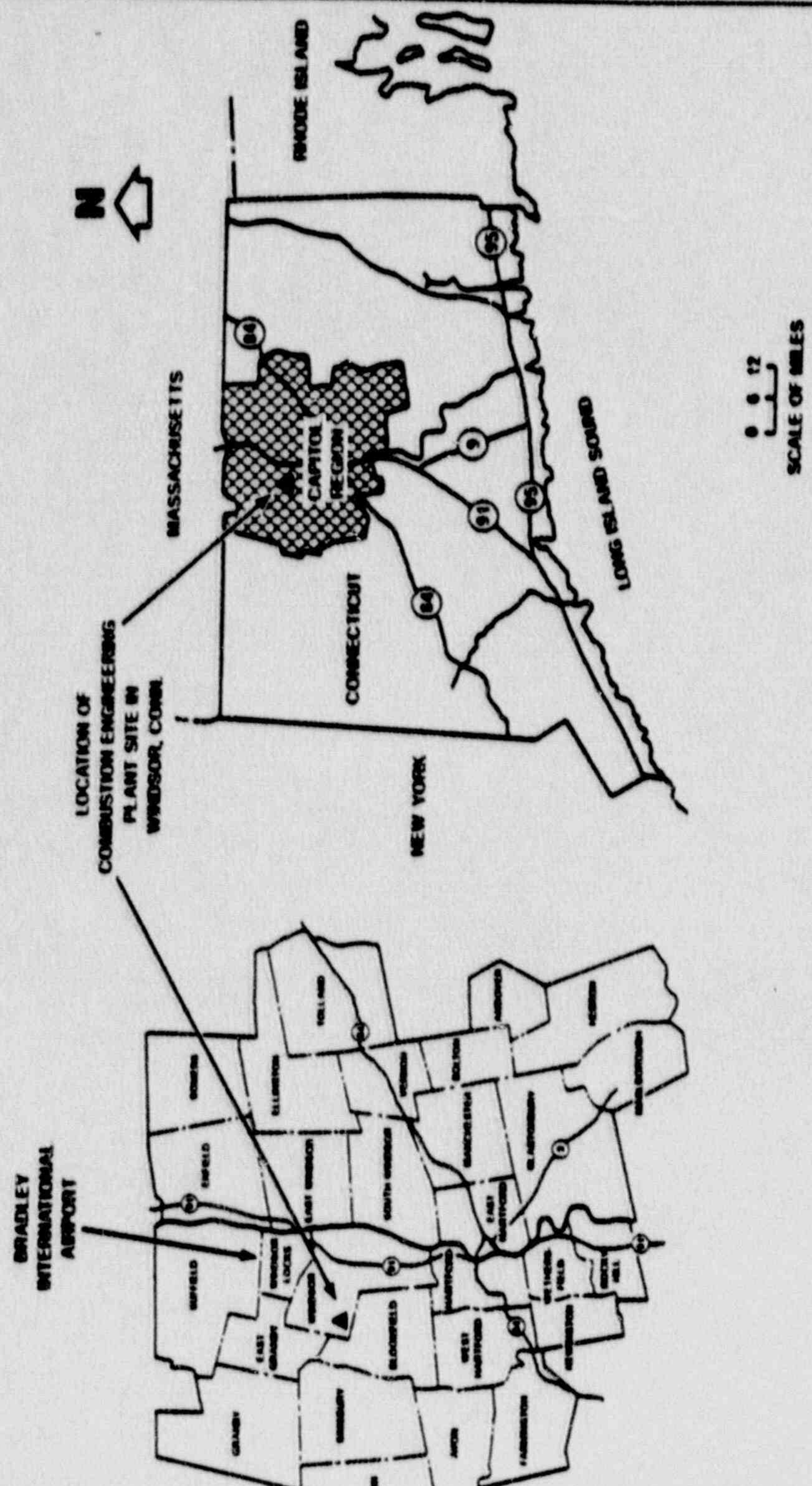


Figure 9-1 General Location Map

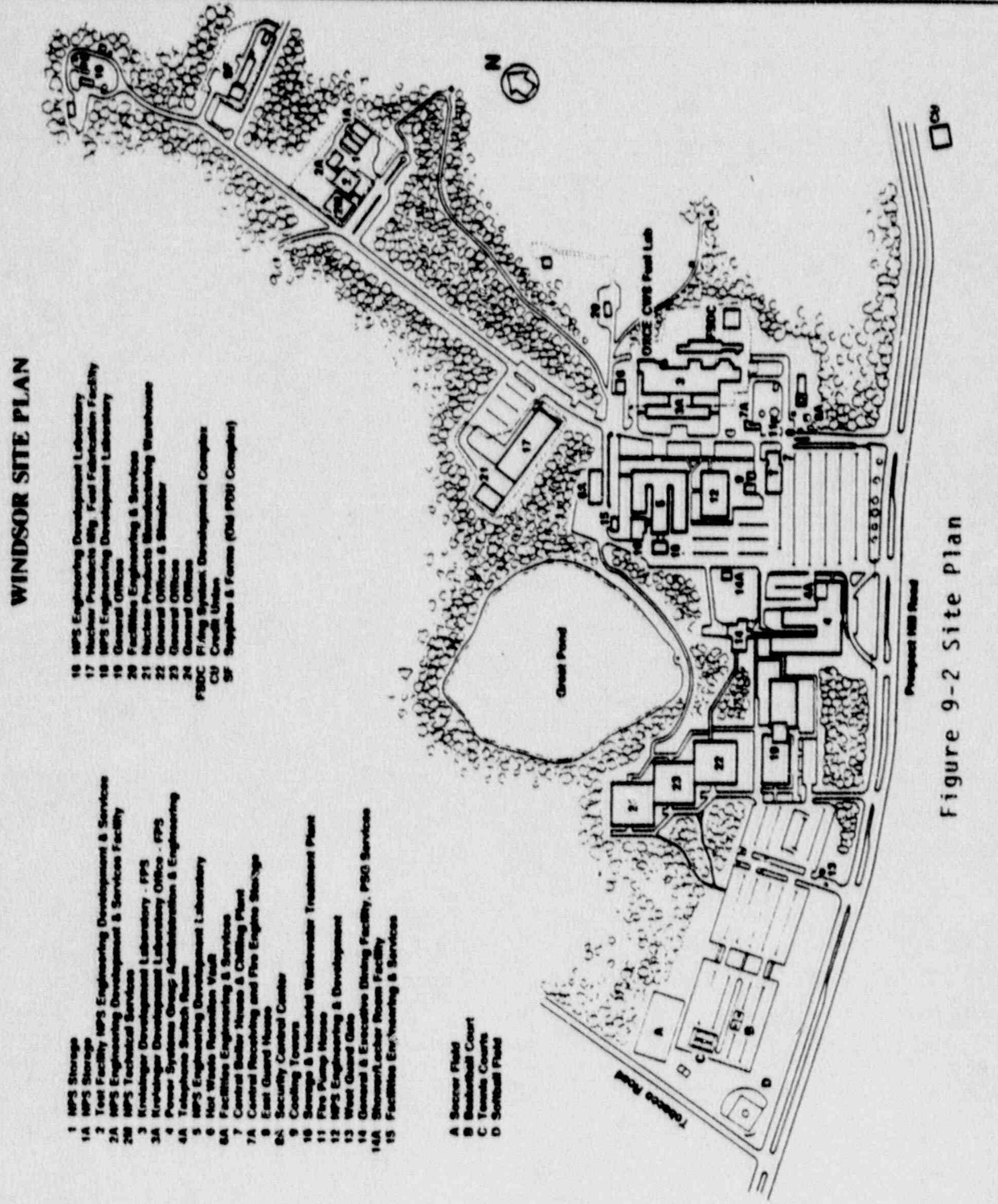


Figure 9-2 Site Plan

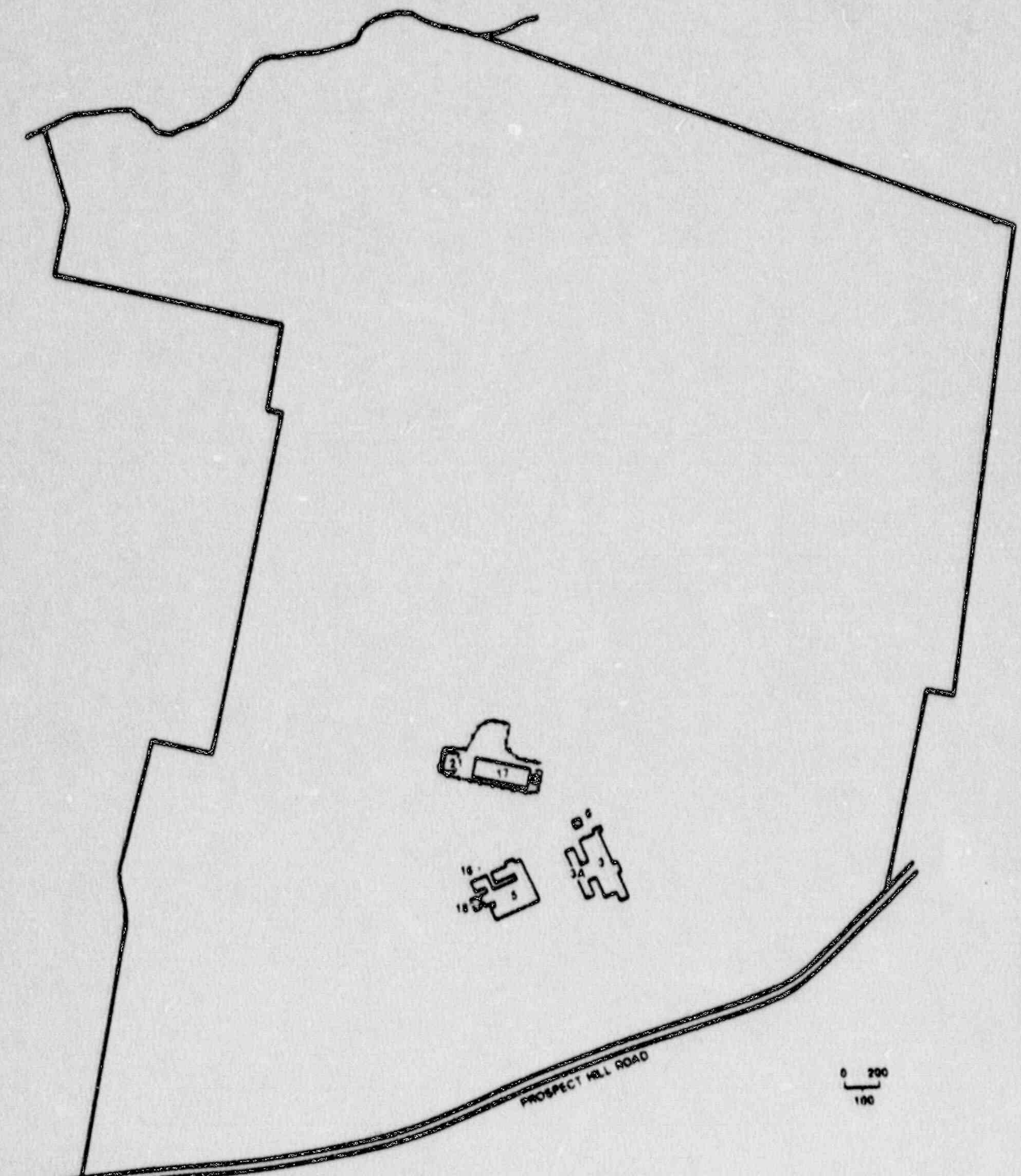


Figure 9-3 Site Boudary

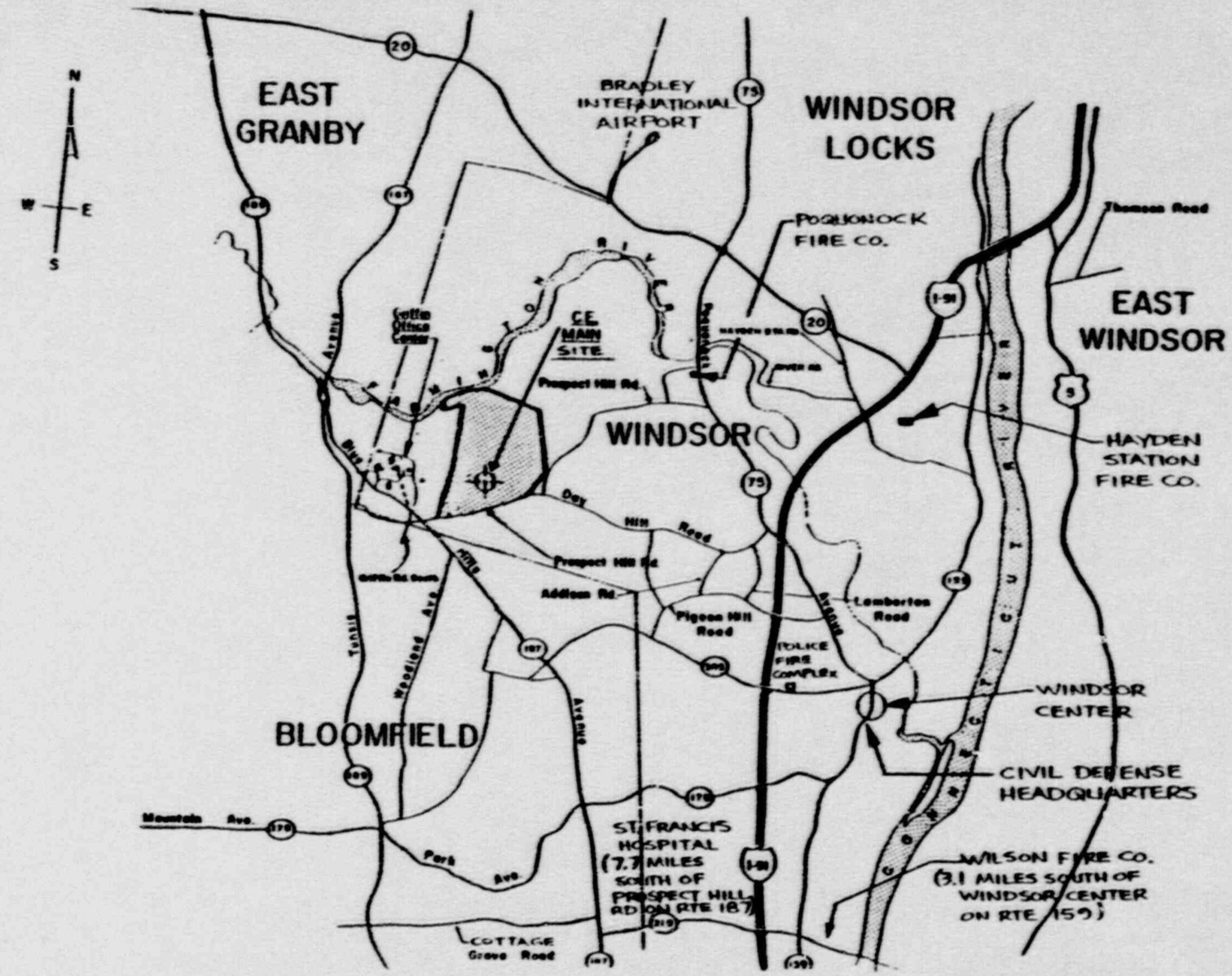


Figure 9-4 Site Vicinity

CHAPTER 10 INTENDED POST-REDEPLOYMENT FACILITY DESCRIPTION

10.1 Plant Layout

Figure 10-1 shows the layout of Building 5. Figure 10-2 shows the layout of the ground floor of Building 17. Fuel fabrication activities are not conducted on the Building 17 mezzanine levels.

In Figure 10-2, portions of Building 17 are lebeled, "Area Being Redeployed". These areas contain equipment that had been used for pellet production. Although this equipment temporarily remains in the facility, it is not presently used in the fuel fabrication process and will be removed as the redeployment program proceeds. This equipment is not shown for these reasons.

10.2 Utilities and Support Systems

10.2.1 Electric Power

The normal source of electrical power is the Northeast Utilities distribution system which provides 22.9 Kv, 60cycle, 3 phase power to a step-down transformer at Building 17. The step-down transformer, rated at 3,750 Kva, provides 480v, 60 cycle, 3 phase power for distribution within the facility. Further voltage step-downs to 208/120v are made to provide power for lighting and general use.

An emergency diesel generator rated at 200 Kw provides 480v, 60 cycle, 3 phase backup power. The generator feeds a power distribution panel that is automatically transferred to the emergency generator upon loss of the normal power source. Power is distributed from the panel through switchgear and step-down transformers which provide power to emergency loads. Lighting and facility alarms are provided with emergency power. The emergency diesel generator is started and loaded weekly to test its operability.

10.2.2 Compressed Air

Compressed air is not used for personnel protective equipment in the fuel fabrication process.

10.2.3 Water

Water is supplied to the site by the Metropolitan District Commission (MDC) through a 12 in. main and distributed to Buildings 17 and 5 as shown in Figure 10-3. The total daily volume supplied to Buildings 17 and 5 is approximately 10,500 gal. for the uses shown in Table 10-1.

The fuel fabrication process does not require water for safety purposes. Loss of the normal water supply is, therefore, not an occurrence having safety implications for normal operations.

With respect to fire protection, water for building sprinkler systems and fire hydrants is provided from a 250,000 gal. ground-level storage tank. The tank receives water from the MDC via the 12 in. main. An electric driven fire pump distributes water from the tank.

A backup supply of water for fire fighting purposes is provided by a 150,000 gal. elevated storage tank. Water for the tank is supplied from the site wells. Diesel driven pumps provide well water to the tank and distribute fire fighting water to building sprinkler systems and fire hydrants if electrical power is lost.

10.3 Ventilation Systems

Three ventilation systems important to safety are provided in Building 17 and are designated as Systems FA-1, FA-3 and FA-4. System FA-2 had served various furnaces associated with pellet production and is no longer used. Each system contains prefilters and a double bank of HEPA filters.

Airborne radioactive material is exhausted from Product Development (Building 5) via five individual stacks. All stacks have absolute filters.

Action levels, monitoring requirements and testing requirements pertinent to meeting regulatory limits are provided in Section 3.2.2.

Disposal of used ventilation system filters is described in Section 10.4.2, Solid Waste.

10.4 Radioactive Waste Handling

10.4.1 Liquid Wastes

Liquid wastes in Building #17 which contain U02 are generated as floor mop water, clean-up water, and water from the sinks and showers in the change rooms. Liquid wastes in Building #5 which contain U02 are generated primarily by wet chemical analysis and cleaning of glassware used in the analysis of U02.

Liquid wastes collected in portions of Building 17 where uncontained fuel material is handled are processed to reduce the amount of U02 transferred to the Building 6 waste retention tanks and, ultimately, to the environment. The wastes are drained into a closed loop slab storage tank/ centrifuge system.

Periodically the tank contents are cycled through the centrifuge and sampled. IF the radioactive content is sufficiently low, the liquid is transferred to the Building 6 liquid waste retention tanks.

Wastes from Building 5 are collected, sampled and released to the retention tanks in Building 6. The waste lines from Buildings 5 and 17 connect in a common line to Building 6.

The wastes drain to any one of ten 2000-gallon retention tanks in Building #6. The tanks fill automatically in sequence. Electronic measuring devices activate a signal when two retention tanks remain in reserve to receive radioactive liquid waste.

Each retention tank is agitated and circulated to provide a representative sample. A sampling station is located at the base of each tank. A sample is withdrawn prior to discharge, and forwarded to the radio-chemistry laboratory for gross alpha and beta analysis. If levels are in excess of 10 percent MPC_W then the waste liquid is transferred to one of four 5000-gallon dilution tanks and diluted as necessary to a maximum concentration of 10 percent MPC_W.

Liquid wastes from Building 6 are discharged via the industrial waste line to the C-E site creek which discharges into the Farmington River.

10.4.2 Radioactive Solid Wastes

Solid wastes containing UO_2 are generated in Buildings #5 and #17 in the form of HEPA filters, rags, paper and other miscellaneous materials generated during normal processing operations. The waste is packaged and either shipped to Combustion Engineering's Hematite, Missouri facility for waste reduction and/or recovery, or transferred directly to a licensed waste processor and/or licensed disposal facility in accordance with applicable requirements specified in Title 10 and Title 49 of the Code of Federal Regulations

Shipping packages containing solid radioactive waste are stored on an outdoor pad contiguous to the south wall of the Building 21 warehouse before transferal to a licensed waste processor and/or licensed disposal facility. The pad is 14'x80' and is in the Building #17 and #21 complex which is enclosed within an 8' high chain link fence. Section 6.4.3 Solid Radioactive Waste, contains requirements that ensure safe storage of the wastes.

In 1988, the low level waste generated in Buildings #5 and #17 for disposal was approximately 3,200 cubic feet. Removal of pellet production from the Windsor site is expected to reduce the quantities of waste produced.

10.5 Fire Protection

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 10-4.

Fire protection, including sprinklers, is designed into all buildings which are subject to fire damage. The manufacturing and laboratory facilities are constructed and operated consistent with requirements of the applicable NFPA fire safety codes.

Table 10-1 Building 5 and 17 Water Utilization

Bldg. No.	Process Water (gpd)	Equipment Cooling Water (gpd)	Water for Sanitary Purposes (gpd)
5	800	2,500	2,000
17	1,200	3,000	1,000

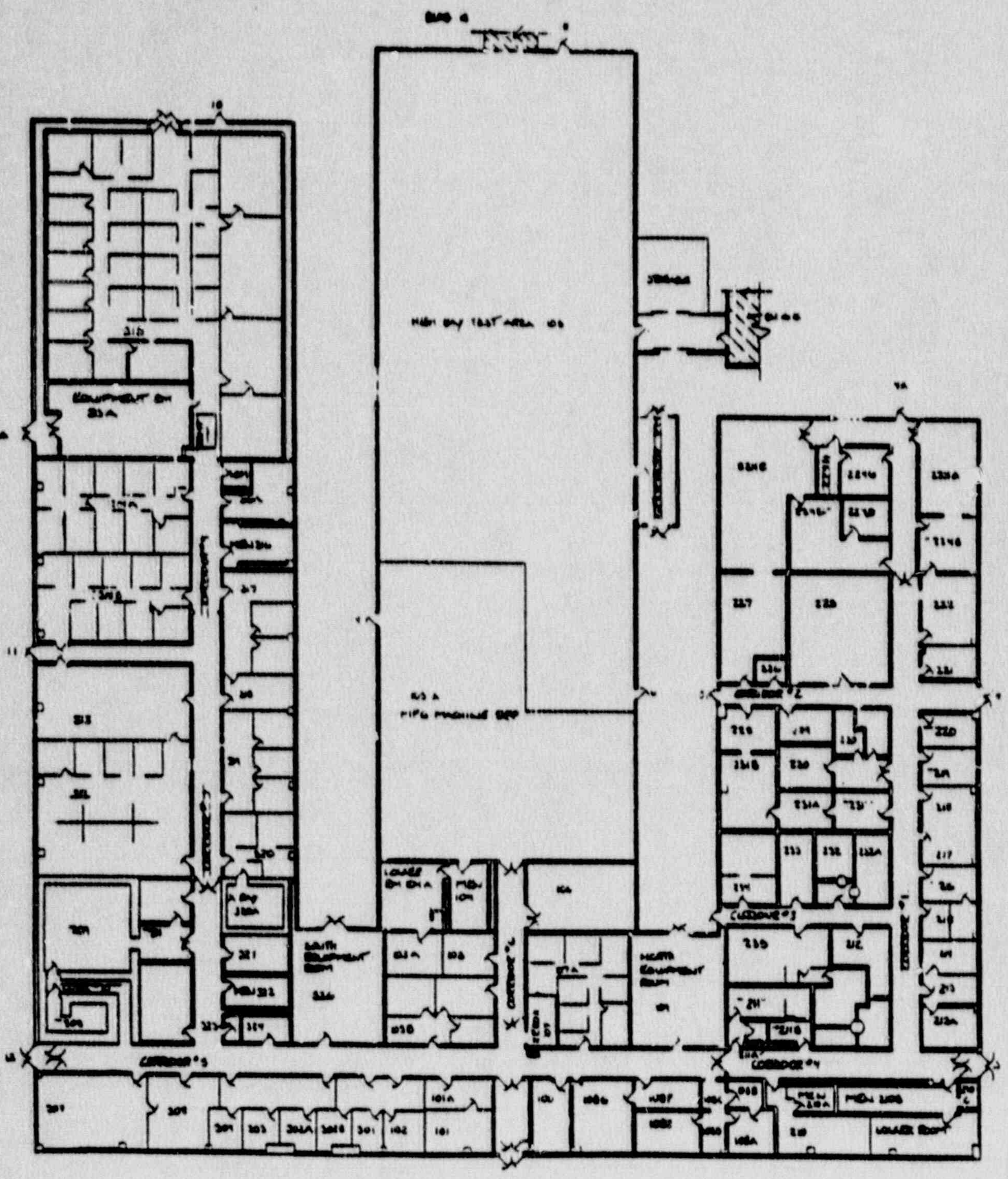


Figure 10-1 Building 5 Floor Plan

OVERSIZE DOCUMENT PAGE PULLED

SEE APERTURE CARDS

NUMBER OF OVERSIZE PAGES FILMED ON APERTURE CARDS 1

APERTURE CARD/HARD COPY AVAILABLE FROM

RECORDS AND REPORTS MANAGEMENT BRANCH

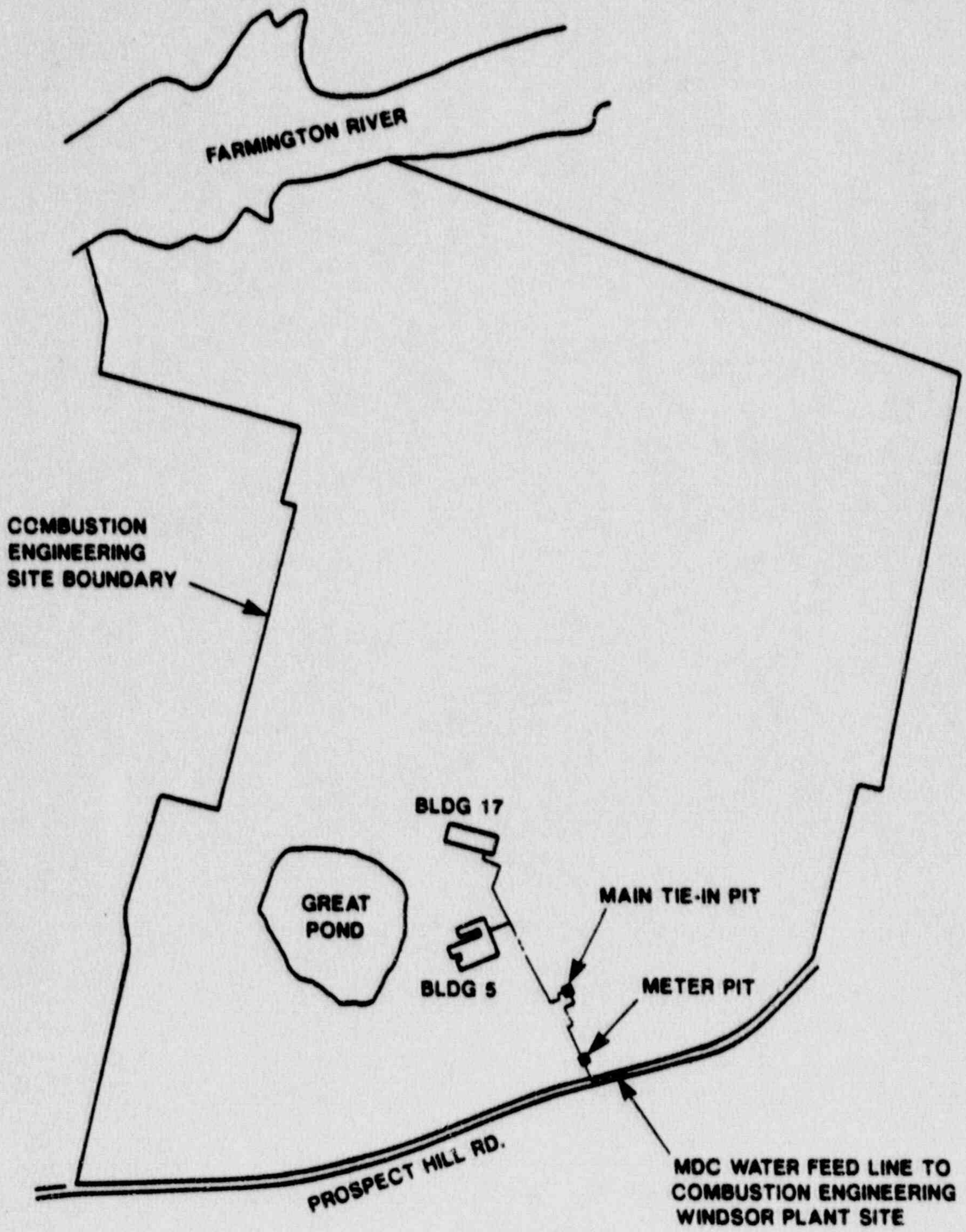


Figure 10-3 Water Distribution To Building 5 And 17

AMERICAN NUCLEAR INSURERS

The Exchange, Suite 245 / 270 Farmington Avenue
Farmington, Connecticut 06032 / (203) 677-7305 / TLX. No. 643-029

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

CERTIFICATE OF INSURANCE

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

Name of Insured: COMBUSTION ENGINEERING, INC.

Mailing Address: 900 Long Ridge Road, Stamford, Connecticut

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

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

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

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

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

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

Effective date of the Certificate: July 1, 1989

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

Figure 10-4 Certificate Of Insurance

CHAPTER 11 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 11-1.

11.1 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 other personnel holding management positions.

11.1.1 Manager, Nuclear Materials Licensing

The Manager, Nuclear Materials Licensing reports to the Director, Nuclear Licensing, who reports functionally to the Vice President, Nuclear Fuel 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.

11.1.2 Manager, Nuclear Materials

The Manager of Nuclear Materials reports to the Vice 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. He 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.

11.1.3 Manager, Quality Assurance

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

11.1.4 Manager, Accountability and Security

The Manager of Accountability and Security reports to the Plant Manager. The implementation of the Fundamental Nuclear Material Control Plan, maintaining custodial control of nuclear materials, warehousing when not under the control of the Production Manager and management of radioactive waste are the responsibility of the Manager of Accountability and Security. He 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.

11.1.5 Operations Supervisors

The Operations Supervisors report to the Plant Manager. He is 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.

11.1.6 Manager, Training

The Manager of Training reports to the Plant Manager. He 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.

11.2 Resumes of Key Personnel Important to Safety

Resumes of key personnel important to safety are provided on Pages 11-4 through 11-17.

PART II
SAFETY DEMONSTRATION

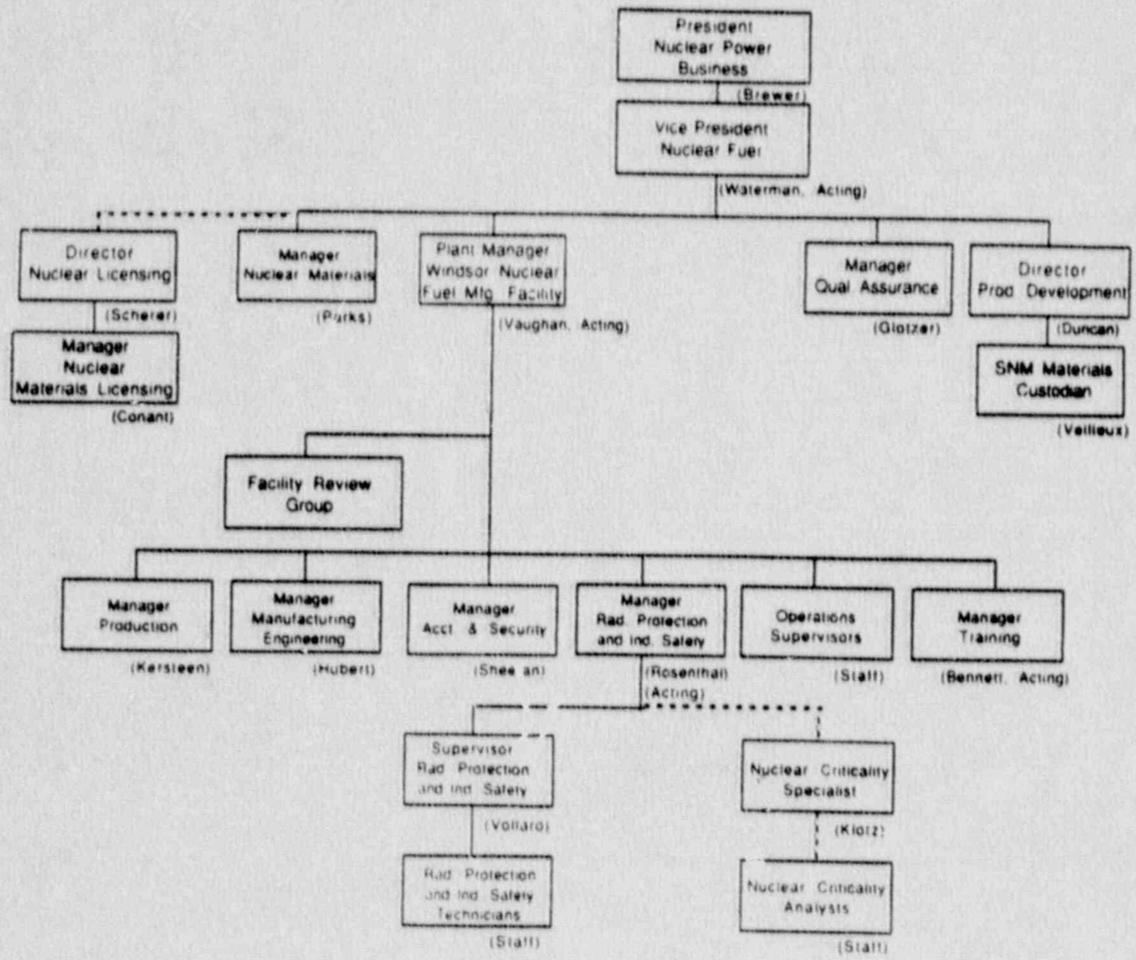


Figure 11-1
WINDSOR FUEL MANUFACTURING FACILITY AND PRODUCT DEVELOPMENT ORGANIZATION STRUCTURE

ROBERT NORMAN DUNCAN - Director, Product Development

EDUCATION

B.S. Metallurgy, Massachusetts Institute of Technology, 1955.
Additional Study, Stanford University, Graduate Materials Science
Courses, 1964-1968
Registered Professional Engineer (California) #MT-1047.

EXPERIENCE

COMBUSTION ENGINEERING, INC.

Director, Product Development 1985 to Present

Responsible for the management of all nuclear fuel product development laboratory activities. This responsibility includes operations, material accountability, security, radiological, criticality and industrial safety, environmental protection, and licensing.

Director, Fuel Development Department 1974 to 1985

Responsible for direction of core materials development including programs to develop and test fuel, cladding and control materials. Functional responsibility included Materials Technology, Fuel Performance Testing and Analysis and Analytical Chemistry.

Responsible for planning and executing experimental programs to improve C-E fuel performance, evaluating and monitoring of fuel performance in C-E supplied plants and technical integration of development activities with KWU.

Manager, Core Materials 1972 to 1974

Responsible for line management of the Core Materials Development organization including planning and execution of PED projects related to fuel testing, examination and performance analysis. Initiated several major fuel irradiation test programs in both test and power reactors. Responsible for project field support, on-site fuel examinations and plant licensing activities related to fuel performance. Responsible for the integration of fuel performance related R&D with the appropriate KWU programs.

ROBERT NORMAN DUNCAN

Staff Assignment Reporting to Director Nuclear Laboratories — May, 1972 to Dec. 1972

In a staff function, reviewed C-E positions on licensing, prepared and reviewed PED programs, participated in early KWU/C-E information exchanges on fuel performance and planned integrated C-E/KWU irradiation test programs. Planned and initiated an experimental irradiation test program on fuel densification and negotiated agreements for joint programs in this area with EEI and others. Evaluated the impact of C-E participation in the Halden Reactor Project and assisted in the negotiations of the agreements.

GENERAL ELECTRIC COMPANY — 1963 TO 1972

Breeder Reactor Development Operation and Nuclear Fuels Development

Managed Fast Reactor Ceramic Fuels and Cladding Metallurgy Units. Directed research and development effort on stainless steel clad uranium-plutonium oxide fuels for fast breeder reactor application, including basic physical property measurements and an extensive irradiation test program conducted in EBR-II.

Responsible for zirconium fuel cladding development program coordination and irradiation testing in the Big Rock Point reactor.

Project Engineer of the AEC-Euratom Specific Zirconium Alloy Design Program.

Project Engineer of the AEC-Euratom Stainless Steel Failure Investigation Program involving stress-assisted intergranular corrosion of austenitic stainless steels in water reactors.

AEC High Power Density Fuels Development Program, including post-irradiation examinations of VBWR fuel assemblies.

WESTINGHOUSE ELECTRIC CORPORATION ATOMIC POWER DEPARTMENT — 1958 to 1963

Senior Engineer

Designed irradiation test program on Zircaloy clad UO2 fuel rods.

ROBERT NORMAN DUNCAN

GENERAL ELECTRIC COMPANY — 1956 to 1958
ATOMIC POWER EQUIPMENT DEPARTMENT

Worked on development of process and fabrication techniques on reactor core materials.

WESTINGHOUSE ELECTRIC CORPORATION — 1955 to 1956

Worked on fabrication techniques for uranium-aluminum fuel.

PHILIP R. ROSENTHAL - Manager, Radiological and Industrial Safety (Acting)

EDUCATION

B.S. Mechanical Engineering, University of Maryland, 1959
Graduate Study, Nuclear Engineering, Drexel Institute of Technology, Philadelphia, Pennsylvania, 1962 - 1963

EXPERIENCE

COMBUSTION ENGINEERING, INC. 1970 to Present

Program Manager, Radiological and Industrial Safety — Jan. 1989 to Present

Responsible for definition of radiological, criticality, and industrial safety and environmental protection programs and standards for Combustion Engineering's Windsor and Hematite Nuclear Fuel Manufacturing facilities. Also responsible for emergency planning at the Windsor facility. Through an audit function, assure compliance and proper implementation of safety related programs and standards.

Manager, Radiological and Industrial Safety — June, 1988 to Jan. 1989

Responsible for the radiological protection, health physics, and industrial safety programs at Combustion Engineering's Windsor Nuclear Fuel Manufacturing facility. Ensured that the necessary engineering and administrative controls were in place for radiological protection, criticality safety, and industrial safety. Reviewed and approved all proposed alterations, modifications, and additions where the use of radioactive materials where involved.

Manager, Radiological Protection Services — 1974 to June, 1988

Responsible for all radiological protection and health physics activities for Combustion Engineering, Inc. Administered NRC licenses for by-product, source, and special nuclear materials. Reviewed and approved all proposed alterations, modifications, and additions to facilities where the use of radioactive material was involved. Responsible for all C-E operations involving radioactive waste disposal, decommissioning, environmental control and the transportation of radioactive materials. Responsible for providing health physics and radioactive material control services for all C-E field service activities. Provided consultation services to utilities in the area of radiological protection, waste disposal, and radioactive material transportation.

PHILIP R. ROSENTHAL

Program manager for development of a computerized dose tracking and evaluation system for power plant maintenance activities.

Responsible for the design, construction, and licensing of a facility for contaminated and irradiated equipment.

Program manager for three decommissioning projects including the decontamination and decommissioning of a uranium fuel fabrication facility to unconditional release criteria.

Supervisor of Mechanical Development 1970 to 1974

Responsible for mechanical development and test facilities for pressurized water reactor systems and components.

GENERAL DYNAMICS CORPORATION
ELECTRIC BOAT DIVISION

Senior Project Engineer 1964 to 1970

Responsible for the design, procurement, test and evaluation of submarine reactor plant components.

Refueling Director

Developed refueling procedures, tools and equipment for refueling submarine reactors. Supervised the refueling of six naval reactor plants.

UNITED STATES NAVY (Civilian Employee)

Mechanical Engineer 1962 to 1964

Responsible for nuclear engineering research and development programs.

MARTIN-MARIETTA CORPORATION

Reactor Systems Engineer 1958 to 1962

Responsible for design and fabrication of a liquid radioactive waste disposal system.

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 providing 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.

ROBERT J. KLOTZ - Nuclear Criticality Specialist

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.

JOSEPH J. VOLLARO - Supervisor, Radiological Protection and Industry Safety

EDUCATION

Hazardous Waste Management Course, satisfies EPA RCRA mandate per 40CFR265.16, October, 1988.

Certified Radiation Safety Officer, January, 1988.

Respiratory Protection at Nuclear Power Plants, by Radiation Safety Associates, 1987.

Nuclear Criticality Short Course, University of New Mexico, May, 1985.

Certified NRRPT, Fall, 1985.

Four year training course as a Dosimeter/Health Physics Technician per NAVSHIPS 0288, General Dynamics Electric Boat Division, 1973 - 1977

Two semesters Physics/Physical Science, Eastern Connecticut State College, 1972 - 1973.

Training Course in "Radiation Protection Technology", Combustion Engineering

College Preparatory, Plainfield High School, 1972

EXPERIENCE

COMBUSTION ENGINEERING, INC. — Nov. 1980 to Present
Windsor, Connecticut

Supervisor, Radiological Protection and Industrial Safety — Nov. 1985 to Present

Provide surveillance of nuclear fuel manufacturing and product development activities for radiological, criticality and industrial safety, environmental protection and emergency planning. Issue Radiation Work Permits for activities outside scope of routine operating procedures or which require enhanced radiological protection.

Senior Health Physics Technician — 1980 to Nov. 1985

Responsible for implementing and monitoring industrial safety, radiation, contamination and criticality control program. Also responsible for packaging/transportation of radioactive waste/hazardous waste program, ALARA and implementing respiratory protection program.

JOSEPH J. VOLLARO

GENERAL DYNAMICS ELECTRIC BOAT DIVISION
Groton, Connecticut

Senior RadCon Monitor 1973 to 1980

Responsible for training of Electric Boat employees in usage of TLD's, radiation control, contamination control, usage of TLD reader/computer interface equipment. Also performed routine/non-routine whole body counting of facility personnel, operated TLD readers and whole body counters, maintained personnel and environmental TLD program, directed monthly/quarterly radiation drills, performed audits of facilities, performed high/low range surveys of reactor shielding during initial start-up.

GARY C. KERSTEEN - Manager, Production

EDUCATION

B.S. Mechanical Engineering, Trinity College, 1968

EXPERIENCE

COMBUSTION ENGINEERING, INC. — 1974 to Present
Windsor, Connecticut

Manager, Production — 1979 to Present

Responsible for all aspects of planning, scheduling, material control, and production control at the Windsor Nuclear Fuel Manufacturing facility. This includes inventory accounting for non-nuclear commodities and the operation of warehousing, receiving and shipping functions. Also responsible for training of facility operations personnel in proper use of operating procedures and adherence to safety limits.

Supervisor, Nuclear material Accountability — 1974 to 1979

Responsible for all aspects of nuclear material control, measurement control, and statistics. Included in the responsibilities of the position was the development and operation of computer system for nuclear material accountanting, item control and measurement control.

MILITARY SERVICE: U. S. ARMY — 1969 to 1974

Commander, 575th Ordnance Co. Germany — 1972 to 1974

Responsible for 150 men assigned to the 575th Ordnance Co. in electronics, mechanical repair, depot supply and support sections. Responsible for general electronic and mechanical repair support to the Sergeant guided missile systems for the European Theatre.

Supply Officer, 575th Ordnance Company — 1970 to 1972

Responsible for 30 men and 8000 line items at a depot level supply warehouse furnishing repair parts to missile units in Europe.

PAUL W. HUBERT - Manager, Manufacturing Engineering

EDUCATION

B.S. Materials Science Engineering, San Jose State College, 1966

EXPERIENCE

COMBUSTION ENGINEERING, INC. — Jan. 1970 to Present
Windsor, Connecticut

Manager, Manufacturing Engineering — 1974 to Present

Direct engineering of the Windsor Nuclear Fuel Manufacturing facility equipment, process, methods and construction for new facilities and for modifications. Prepare procedures, perform training and maintenance for facility processes and equipment. Verify that facility engineering satisfies requirements on criticality, radiological and industrial safety and environmental protection.

Engineering Supervisor, Fuel Pellet Fabrication Facility — 1970 to 1974

Responsible for all engineering activities associated with the Windsor Fuel Manufacturing Facility. Responsible for plant layout, equipment specification, process development, plant startup, manufacturing procedures, material specification, and production engineering.

UNITED NUCLEAR CORPORATION

Project Engineering, Semi-works Oxide and Pellet Plant — 1966 to 1970

Supervised the activities of an engineering group engaged in designing, building, starting up, and operation of two UO_2 pellet production lines. Responsibilities included plant layout, equipment and materials specification, process development, and plant start-up.

GENERAL ELECTRIC COMPANY

Manufacturing Engineering, Fuel Production Engineering — 1958 to 1966

Responsible for equipment and materials specification, manufacturing procedures, and process specification for production of UO_2 fuel pellets, fuel rods, and other nuclear fuel components. Prior to this assignment, held various technical positions in Fuel Development Engineering, Fuel Design Engineering, and Quality Control.

RAYMOND E. VAUGHAN - Plant Manager (Acting)

EDUCATION

M.S., Systems Management, University of Southern California, 1975
B.S., Marine Engineering, U.S. Naval Academy, 1963

Supplemental Education:

Naval Submarine Prospective Commanding Officer School, 1978
Naval Submarine and Nuclear Power School, 1965

EXPERIENCE

COMBUSTION ENGINEERING, INC.

Plant Manager (Acting) Dec. 1989 to Present

Manager, Operation - Windsor Nuclear Fuel Manufacturing Facility 1989 to Dec. 1989

Responsible for overall coordination of activities amongst all Line Managers to ensure that the facility production goals are satisfied within the limits imposed by Federal, State and local regulations. Also serve as Emergency Director for nuclear fuel manufacturing facility and product development laboratories emergency response team.

Nuclear Fuels Independent Task Force Aug. 1988 to 1989

Assigned to an eleven-member Independent Task Force established to conduct an audit of C-E's Windsor Nuclear Fuel Manufacturing Facility. Audit results determined the status of compliance with applicable license conditions and regulatory requirements. Areas of review included; manufacturing operations, nuclear criticality safety, radiological controls, health physics, industrial and environmental safety, maintenance, licensing, training, emergency preparedness, and organizational and management effectiveness.

Manager, Nuclear Startup 1983 to 1988

Responsible for the organization and direction of a multi-disciplinary engineering staff providing a wide range of engineering services at nuclear power plants. Directed the establishment and administration of the C-E Site Startup offices at projects in which the Nuclear Steam Supply System was provided by C-E.

RAYMOND E. VAUGHAN

Project Manager, Technology Transfer, Korea — Sept. 1986 to Mar 1987

Assigned, on loan, to the NSSS Projects Department throughout the contract negotiation phase for Korea Nuclear Units 11 and 12. Responsible for preparing, resolving and coordinating all technical, commercial and legal terms leading to award of two Technology Transfer Agreements to provide technology to Korea Heavy Industries and Construction, Inc. for NSSS Component Design and Manufacturing; and Korea Advanced Energy Research Institute with NSSS System Design and Fuel/Core Design.

UNITED STATES NAVY

Anti-submarine Warfare and Submarine Liason Officer, Staff, Cruiser-Destroyer Group Five	1983
Deputy, Training and Readiness, Staff, Submarine Squadron Three	1982
Commanding Officer, USS GUARDFISH (SSN-612)	1979 to 1982
Executive Officer, USS SEADRAGON (SSN-584)	1975 to 1979
Director, Tactical Training Department, Naval Submarine Training Center, Pacific	1973 to 1975
Engineer Officer, USS FLASHER (SSN-613)	1969 to 1973
Weapons Officer and Main Propulsion Officer, USS HADDOCK (SSN-621)	1966 to 1969
Assistant Engineer and Sonar/Communication/ Electronics Material Officer, USS CHOPPER (SS-342)	1965 to 1966

Technical Naval Experience

Power Plant Maintenance and Operation

Extensive experience in new construction, overhaul, refueling, testing, maintenance, inspection, training and operation of naval nuclear propulsion power plants, including pre-commissioning and overhaul crews in private and naval shipyards. Qualified as an Engineering Officer of the Watch on four different reactor plant designs. Certified by Naval Reactors as an Engineer Officer qualified for supervision of maintenance and operation of naval nuclear power plants.

RAYMOND E. VAUGHAN

Weapons and Integrated Logistics Systems Management

In-depth experience in theory, operation and maintenance of nuclear power plants, complex electronics systems, weapons, sensor and communications systems.

Project Management

Overall manager for development and installation of advanced submarine (SSN-688) update to Integrated Submarine Tactical Team Training Device 21A38; and for Advanced Submarine ASW Training Device 21A41 Proposed Military Characteristics design and procurement specifications.

CHAPTER 12 INTENDED POST-REDEPLOYMENT RADIATION PROTECTION

12.1 Program

Combustion Engineering maintains a radiation protection program based upon written program documents and written implementing procedures. Subsequent sections of this chapter describe the conduct of radiation surveys, ALARA measures and methods for monitoring personnel exposure and the contamination of equipment and surfaces including actions taken to control exposures and contamination.

12.2 Posting and Labeling

Any area inside a restricted area to which access is controlled for the purpose of protecting individuals from exposure to radiation or radioactive materials is called a radiological control area. The radiological control areas are of the following types:

1. Radiation Area
2. Airborne Radiation Area
3. Contaminated Area
4. Radioactive Materials Area

The posting, labeling, and control within these areas is in accordance with 10 CFR 20.203. All posting and labeling will be conspicuously displayed.

12.3 External Radiation - Personnel Monitoring

The objective of the personnel monitoring program for external radiation is to ensure that the radiation exposure received by Combustion Engineering employees and visitors at the Windsor Nuclear Fuel Facility shall not exceed the NRC limits as stated in 10 CFR 20.

10 CFR 20.202(a) requires that appropriate personnel monitoring equipment be provided to each individual who enters a radioactive material area where he will receive, or is likely to receive, a dose in a calendar quarter in excess of 25 percent of the NRC limit permitted in 10 CFR 101(a).

C-E issues a beta/gamma sensitive dosimeter (TLD) and an indium foil to each worker entering a radiological control area. The range of the TLD is from zero to 99.99 rem. All visitors, i.e., any individual seeking entry into radiological control areas who has not successfully completed the Radiation Workers Training Program, is issued the neutron foil badge. Visitors must be escorted and may only observe.

Other types of dosimeters such as finger rings or wrist bands may be used, in addition, where special surveys or situations require specific information. Self-reading dosimeters may also be used where the potential for higher than expected exposures exists.

For employees working in radiological control areas where there is the potential of exceeding 25% of the quarterly NRC limit, the TLD's are analyzed at least quarterly. The exposed TLD's are processed by a NAVLAP certified laboratory for analysis.

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. In the case of a criticality accident, these badges and foils would be processed immediately. The procedures to be followed to determine the high radiation doses immediately following a criticality accident are described in the Emergency Plan Implementing Procedures.

Dosimetry results are evaluated by the Radiological Protection and Industrial Safety Staff to determine that the external exposures received by the workers at the Windsor Fuel Facility are within NRC limits and that no unexpected exposure levels or trends are being exhibited. This external exposure data is also reviewed as part of the ALARA Program to assess the effectiveness of all programs and procedures in maintaining exposures and radiation levels as low as reasonably achievable.

12.4 Radiation Surveys

Routine surveys determine the radiation and contamination levels at various locations at regular intervals. This information permits identifying changed conditions or trends that require action.

The frequency of routine surveys depends on many factors and is established for the specific operations and facilities involved. In uranium receiving, shipping, inspection and storage areas, external radiation and removable surface contamination surveys are conducted monthly. In active processing, decontamination and change room areas, external radiation surveys are performed monthly while contamination surveys are conducted weekly.

Radiation surveys are also conducted on all items being released from a controlled area to an uncontrolled area. Similarly, surveys are performed on all incoming and outgoing shipments of radioactive material from the site to assure that the shipment is within the acceptable limits established by the applicable federal and local government regulations.

Special radiation surveys are performed, when necessary, to determine the radiation levels for maintenance and other activities. Surveys taken prior to the start of work identify the radiological work environment so that appropriate controls can be implemented. Periodic surveys during the work permit determining whether the controls are effective and continue to be appropriate.

The results of all surveys will be documented and maintained in accordance with the requirements of 10 CFR 20.

12.5 Reports and Records

All reports and records required in Section 2.9 of Part I on Radiation Protection shall be maintained in company files in accordance with the recordkeeping requirements of 10CFR20.401. Such reports and records shall be retained for at least the minimum period required by the regulations or for longer retention periods as may be specified in Part I of this license application.

12.6 Instruments

Combustion Engineering maintains a compliment of radiation measurement instruments sufficient to support operational needs and to conduct measurements necessary to meet regulatory requirements. Table 12-1 provides information on the instruments used.

The Manager of Radiological Protection and Industrial Safety is responsible for the radiation measurement instruments and related equipment. This responsibility includes the calibration and maintenance of the instruments and ensuring that adequate quantities of each type of instrument are available for use. The following general requirements apply to maintaining the radiation measurement instrumentation in calibration.

1. All instruments shall be calibrated to the requirements of Section 1.11 of USNRC Regulatory Guide 8.24, "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication."

2. A member of the Radiological Protection and Industrial Safety staff shall be assigned responsibility for the tracking system for the calibration records of each instrument.

3. Each week the system records shall be reviewed to identify those instruments whose calibration expires in the near future.

4. After calibration, each instrument shall have an identifying tag or sticker which shall identify, as a minimum, the last day of valid calibration.

5. The calibration system shall be organized such that not all the instruments are due for calibration at the same time.

6. Instruments are inspected and calibrated semi-annually and following any maintenance.

7. Source response checks and battery checks are performed on survey instruments daily.

8. Radioactive sources used for calibration are traceable to the National Bureau of Standards.

12.7 Protective Clothing

Protective clothing (PC's) are various special garments provided for use in radiologically contaminated areas which protect the wearer from skin contamination and to minimize the spread of contamination.

Combustion Engineerings' Radiological Protection Instructions specify the PC requirement for normal operation and maintenance assignments. For maintenance tasks or special activities requiring a Radiation Work Permit (RWP), the RWP will specify the requirements for protective clothing and other radiological and industrial safety devices.

Instructions and practice in donning and removing PC's is part of the Radiation Workers Training Program which must be successfully completed by all workers required to wear PC's.

12.8 Administrative Control Levels (Including Effluent Control)

The action levels, alarm set points, frequency of measurements and action to be taken are discussed for the following radiation protection monitoring programs.

12.8.1 Occupational Exposure (Internal and External)

The bioassay program for determining internal exposure is described in Section 12.12. The program follows the guidelines provided in Regulatory Guide 8.11, "Applications of Bioassay for Uranium." 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. The personnel monitoring program for external radiation was described earlier in Section 12.3.

The Manager, Radiological Protection and Industrial Safety is responsible for reviewing the urinalysis and in-vivo results, the dosimeter results, and the radiation survey measurements to determine personnel exposure status, unusual trends or potential exposures, and to advise of any special controls or restrictions that should be applied. If the exposure status of an individual becomes uncertain, that individual may be removed from further exposure.

12.8.2 Airborne Activity

Ventilation of the Nuclear Fuel Manufacturing Facility (Building 17) is provided by three separate exhaust systems. The proper operation of these systems is vital to the overall safe operation of the facility since they control and direct the flow of particulate radioactive uranium away from the workers and the clean, non-restricted areas of the plant.

Specific instructions set forth the monitoring requirements of the ventilation system and establish safe radiological protection limits for operation. Also specified are the corrective actions to be taken to ensure that regulatory requirements and/or license conditions are not exceeded.

The action levels specified include a shutdown requirement for any stack whose sample exceeds 1×10^{-12} lCi/ml. Planned discharges of radioactive material to the general environment from the fuel manufacturing operations and radiation from those operations are maintained well within the annual dose equivalent limits permitted by 40 CFR 190.10.

12.8.3 Liquid Activity

Liquid waste is generated in Building 17 and Building 5 by normal manufacturing processes and laboratory processes. Prior to discharge, all liquid waste is sampled and counted to determine its level of contamination. If the water to be discharged is not at or below 3×10^{-6} lCi/ml for insoluble natural uranium and 3×10^{-9} lCi/ml for unidentified mixtures of radionuclides, the water is diluted before being discharged. (Each of these limits is ten percent of its MPC).

12.8.4 Surface Contamination

Routine and special radiation surveys were discussed earlier in Section 12.4. The action levels and the action to be taken for removable alpha contamination in radiologically and non-radiologically controlled areas are discussed in Section 3.2.6. Leak testing sealed sources, and the survey and decontamination of equipment and facilities prior to release for unrestricted use are also discussed in Section 3.2.6.

12.9 Respiratory Protection

Combustion Engineering's respiratory protection program establishes the requirements for limiting the inhalation of airborne radioactive materials, hazardous chemicals, and/or oxygen deficient air by personnel performing activities covered by the SNM-1067 License. It provides the circumstances under which respiratory protective equipment is required, and the methods for properly selecting and using it.

The preferred method for limiting intake of radioactive materials is through the use of permanently engineered safety features and the use of process, containment, and ventilation techniques. Respiratory protective equipment is used when such measures are not feasible or sufficient.

Respiratory protective equipment is required when it is necessary to reduce the average concentration of radioactive material inhaled to values less than those specified in 10CFR20, Appendix B, Table 1, Column 1, "Concentrations in Air and Water Above Natural Background." Respiratory protective equipment may also be required during emergency situations such as fires and chemical releases, and when entering areas of unknown oxygen content.

C-E's respiratory protection program meets the requirements of several Federal and State government regulations, standards and guides including NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials". The program includes requirements for medical screening, respirator user training, qualitative fit testing, maintenance, and routine testing.

12.10 Occupational Exposure Analysis

Results of a study of exposure records for NFM personnel who were monitored for whole body exposure over approximately 3 1/2 years were reported to cognizant NFM management in November, 1989. The study revealed that the maximum whole body exposure is about 14% of the limit specified in 10CFR20.101(a).

The pellet stacking and rod loading areas are the largest potential source of radiation exposure to the extremeties. A dosimetry study performed in July, 1989 revealed that the dose to the root of the finger and wrist is 14% and 10%, respectively, of the limit specified in 10CFR20.101(a).

12.11 Measures Taken to Implement ALARA

Combustion Engineering's ALARA commitment is stated in Section 3.1.1. In addition to implementation of this commitment through line management, the Facility Review Group's responsibilities include periodic evaluation of Facility operations to detect circumstances or trends that may indicate departures from ALARA practices. The Facility Review Group's findings are reported to the Plant Manager.

12.12 Bioassay Program

The Bioassay Program is used to determine the kind, quantity or concentration of radioactive material found in the human body by direct (in-vivo) measurements or by analysis of materials excreted by the body. The program was developed based on the following documents:

1. 10CFR20, "Standards for Protection Against Radiation"
2. Regulatory Guide 8.11, "Application of Bioassay for Uranium"
3. ICRP-30, "Limits for Intakes of Radionuclides by Workers" (1979)
4. NUREG/CR-4884, "Interpretation of Bioassay Measurements"

The administrator of the Bioassay Program is the Radiological Protection and Industrial Safety Supervisor.

12.12.1 Frequency of Measurement

Individuals who work with uncontained fuel material shall receive semi-annual in-vivo counts and shall also have a urinalysis performed semi-annually. A urinalysis will be performed monthly for all personnel who routinely work with soluble compounds of uranium. More frequent bioassays are performed as required for a sample of the more highly exposed workers as a check on the air sampling program.

12.12.2 Action Limits

A maximum permissible lung burden (MPLB) has been established at 200/1 grams U-235. This value is based on recommendations of the International Commission on Radiological Protection using ICRP-30 and an enrichment of 3.03% U-235. Urine retention fractions are based upon NUREG/CR-4884, "Interpretation of Bioassay Measurements.

1. In-vivo Results Action Levels

In-vivo Result	Action
≥ 200/1 grams U-235	a) Remove individual from further exposure b) Confirm result c) If confirmed: - Impose or continue work limitations - Conduct job investigation - Initiate corrective actions
≥ 175/1 grams U-235	a) Confirm results b) If confirmed: - Impose or continue work limitations - Conduct job investigation - Initiate corrective action

2. Urinalysis Results Action Levels

Sample ≥ 10/1 gram U/liter	a) Confirm results b) If confirmed, the Manager of RPIS will investigate
Sample > 25/1 gram U/liter	a) Confirm results b) If confirmed: - Impose work restrictions - Perform fecal analysis - Collect and evaluate diagnostic urine samples - Conduct investigation - Determine U235 equivalent in lung

In identifying the probable cause of exposure for the above results, the Manager of Radiological Protection and Industrial Safety (RPIS) considers the workers' job assignment, the airborne contamination and adequacy of air sampling, exposure to others, and other bioassay results. The results of all bioassay measurements shall be retained in accordance with the NRC's required record retention period. The Manager of RPIS reviews the bioassay results for potential overexposure problems and trending. He also periodically reviews the plant operations to determine the effectiveness of the Bioassay Program.

12.13 Air Sampling and Internal Exposure Program

Internal exposure monitoring is performed by measuring the amount of radioactivity in the air an individual breathes. The air in all areas where uncontained radioactive materials are handled is monitored as required in 10CFR20.103 to ensure the airborne concentration of radioactivity does not exceed the MPC as specified in 10CFR20, Appendix B, Table 1, Column 1.

There are three types of air monitoring devices currently in use: fixed position samplers, general area samplers and a continuous air monitor. The first two types function in the same basic way, but they serve two different purposes, while the third is different altogether.

The fixed position samplers are intended to monitor the airborne radioactive material concentration associated with a specific task at a specific location. Data collected by these samplers are used to calculate internal exposures for individual who work with uncontained fuel material if the representativeness of the fixed position samples is validated as described in Section 3.2.3. The data may also be used to provide information about the amounts of airborne contamination created by a specific operation and/or the effectiveness of the engineered features used to contain the airborne contamination.

The general area air samplers are used to monitor the airborne concentration in general areas, particularly areas in which a high airborne concentration could be created rapidly by an equipment malfunction. Data collected by these samplers provides approximate average airborne concentrations and are used in trending the information. This information is used to monitor the performance of the ventilation systems and the effectiveness of the contamination control procedures.

The continuous air monitor is a self-contained unit with a filter, a strip-chart recorder, and a built-in alarm function (bell and rotating beacon). Its primary purpose is to provide immediate indication of high airborne radioactive material concentration, so that appropriate actions may be taken to keep worker exposure as low as reasonably achievable (ALARA). The strip-chart provides trend data on the airborne concentration in the vicinity of the monitor.

In addition to the above monitoring devices, lapel air samplers may be used by the individuals who work with or handle uncontained radioactive material, or may be required on other tasks as may be defined by the Radiological Protection and Industrial Safety staff.

All samples from fixed position and lapel air samplers shall be analyzed after each working shift. Suspected contamination of a lapel air sampler filter in a way not representative of an individual's breathing zone requires immediate attention. If a lapel air sampler or fixed position air sampling station indicates a work area with an airborne concentration that exceeds the MPC, the Radiological Protection and Industrial Safety staff shall conduct an investigation as soon as possible. This investigation may include visual inspection, radiation surveys, interviews with workers, and review of internal exposure results of workers in the general area. When the cause of the high airborne concentration has been determined the necessary actions will be taken to reduce the levels to acceptable values and/or to provide for adequate protection for the workers.

12.14 Surface Contamination

Figure 10-2 shows the layout of the ground floor of Building 17. The portion of the building considered to be a contamination area is comprised of the area undergoing redeployment, the pellet receiving and storage areas, the pellet stacking and loading room, the areas containing the pellet drying furnace and end cap welders and the men's and women's change facilities.

Personnel access to the contaminated area is through the change facilities. Personnel monitoring equipment is located in the change facilities at the boundary between the controlled and uncontrolled areas. The boundary is indicated by a line on the floor.

Combustion Engineering's practice on the use of protective clothing is described in Section 3.2.1.3, Protective Clothing. Surface contamination surveys, including frequency and action levels for various areas, are discussed in Section 3.2.6, Surface Contamination. Personnel contamination control is discussed in Section 3.2.1.4, Personnel Monitoring Systems and Section 3.2.1.5, Personnel Decontamination Policy. Table 12-1 provides information on the scintillation counter, auto counter and friskers used for surface contamination surveys and personnel monitoring.

Decontamination of equipment and materials for release for unrestricted use is performed in accordance with USNRC Annex B dated August, 1987.

Table 12-1
Radiation Instruments

Type	Number	Radiation Detected	Range
Scint. Counter	6	alpha	NA
Auto Counter	2	alpha/beta	NA
Frisker	3	alpha	0-200K CPM
Frisker	25	alpha	0-500K CPM
Survey Meter	1	beta/gamma	0-50K CPM
Survey Meter	4	beta/gamma	0-50 MR/HR
Survey Meter	2	beta/gamma	0-200 MR/HR
Survey Meter	1	beta/gamma	0-50 R/HR
Survey Meter	1	beta/gamma	0-500 MR/HR
Survey Meter	1	beta/gamma	0-199 R/HR
Survey Meter	1	beta/gamma	0-200 MR/HR
Continuous Monitor	2	alpha	0-10,000 CPM
Crit. Detector	22	(Note 1)	0-10,000 R/HR

1. Meet 10CFR70.24(a) requirements.

Chapter 13 ENVIRONMENTAL SAFETY - - RADIOLOGICAL

Combustion Engineering's environmental monitoring program is implemented by the Radio-chemistry Staff. Table 13-1 presents the essential features of the program including the types of samples taken, sampling frequency, sampling locations, analyses performed and sample volumes. Figures 13-1 and 13-2, respectively, show the on-site and off-site monitoring locations. Environmental samples have shown no increasing trend in uranium concentration.

Windsor fuel manufacturing operations previously included pellet production and UO2 scrap recycle, activities responsible for a significant fraction of past releases from the facility. Removal of pellet production and UO2 scrap recycle from the Windsor site is expected to reduce the radioactive material released, but operating data on which to base off-site exposure calculations in the absence of these processes is not yet available. The small calculated exposures associated with past operations are expected to be conservative relative to exposures associated with future operations that will not involve pellet production and UO2 scrap recycle. Analyses using past release data resulted in a calculated annual inhalation dose of 0.05 mrem. to an individual at the nearest point on the site boundary to Building 17. The annual inhalation dose to an individual 3 miles away in East Granby, CT (i.e., in the nearest town center to the site) was calculated to be 1.6×10^{-4} mrem. Future operations are expected to result in even smaller calculated doses for the reasons presented above.

Table 13-1 Required Sample Schedule

Sample	Frequency	Location	Analysis	Volume
1. Surface water from Farmington River, Industrial Stream and Site Ponds	Quarterly	Four locations on the Farmington River, the site ponds and industrial stream	Gross Alpha and Beta, Nitrate, Floride, pH, Total Uranium	1.25 liters ea.
2. Well Water	Quarterly	Each site well	Gross Alpha and Beta, Nitrate, pH, Flouride Total Uranium	1.25 liters ea.
3. Sediment from Farmington River, Site Ponds and Industrial Streams	Quarterly	Same locations as surface water	Gross Alpha and Beta, Total Uranium	One pint ea.
4. Vegetation				
On-site	Semi-annually	Each Fallout Station Location and and four locations in grassy area surrounding Building #17	Gross Alpha and Beta, Total Uranium	One pint of packaged vegetation, each sampling station
Off-Site	Semi-annually	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 from each sampling station
5. Soil	Semi-annually	Same locations as vegetation	Gross Alpha and Beta, Total Uranium	One pint (Upper inch) from each sampling station
6. Fallout	Quarterly	Ten locations on-site	Gross Alpha and Beta, Total Uranium	Total Continuous collection at each station

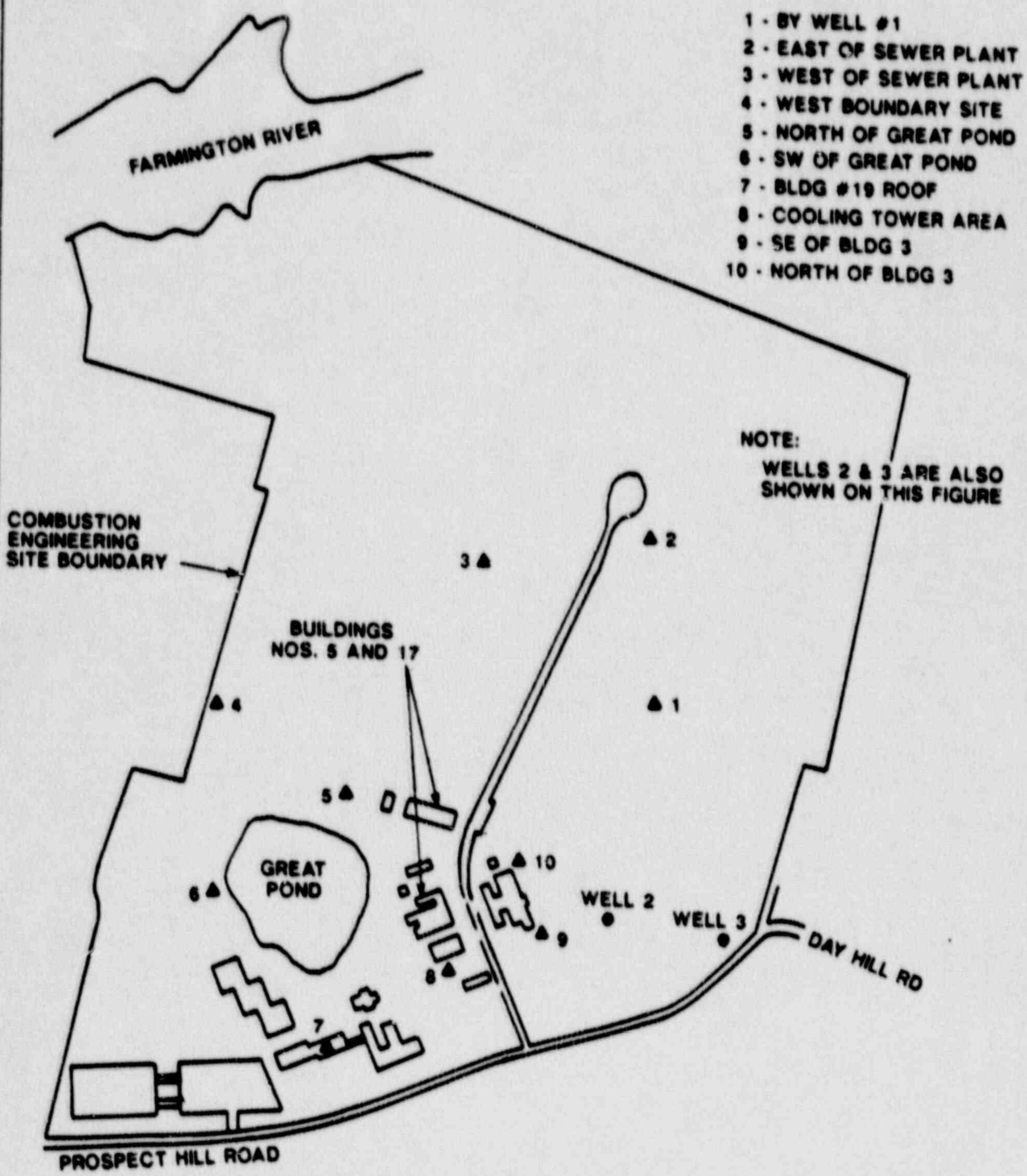


Figure 13-1 Fallout Station And Well Locations

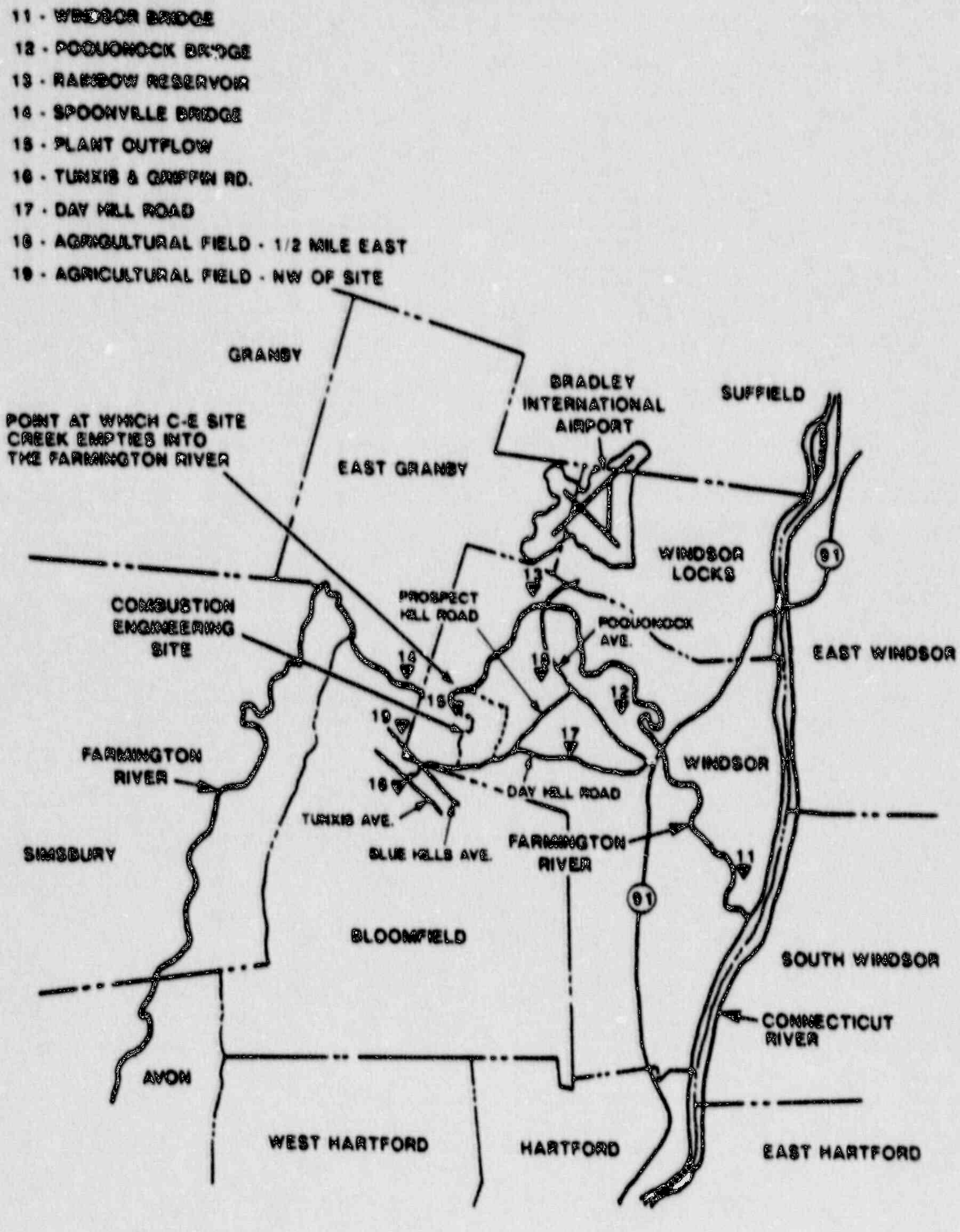


Figure 13-2 Offsite Sampling Locations

CHAPTER 14 INTENDED POST REDEPLOYMENT NUCLEAR CRITICALITY SAFETY

14.1 Administrative and Technical Procedures

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

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

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

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

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

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

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

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

14.2 Preferred Approach to Process Design

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

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

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

Subsequent sections examine methods of control other than safe geometry.

14.3 Basic Assumptions

14.3.1 Analytic Models

14.3.1.1 Individual Units

14.3.1.1.1 Safe Individual Units

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

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

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

Mass	2.3
Volume	1.3
Cylinder Diameter	1.1
Slab Thickness	1.2

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

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

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

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

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

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

14.3.1.1.2 Other Subcritical Individual Units

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

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

14.3.1.2 Nuclear Interaction Methods

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

14.3.1.2.1 Surface Density Model

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

The concept of fraction critical for a SIU is based on a somewhat arbitrary definition deduced from correlations of experimental data; see, for example, Reference 4. The definition employed here takes the fraction critical as the ratio of the SIU mass, or equivalent spherical mass, to that of an unreflected critical sphere of the same composition.

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

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

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

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

For convenience, unreflected extrapolation lengths, λ, are taken from Figure 2 of Reference 4.

λs = 2.1 cm
λt = 2.7 cm
λc = 2.25 cm

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

Figures 14-1 through 14-3 and 14-7 through 14-9 each have a broken line curve showing the dependence of the appropriate fraction critical value versus enrichment; these curves can be compared against the curves showing the minimum critical reflected values reduced by the safety factors to determine which value is more limiting and appropriate for use in the surface density method. For mass limited SIUs, the application of the safety factor to the minimum critical reflected mass is more limiting for both homogeneous and heterogeneous data. For the minimum critical volume parameter, the value deduced using the fraction critical is more limiting except above the 4.5 w/o enrichment value in the heterogeneous UO2 - H2O data. For minimum critical cylinder diameters, values deduced from the fraction critical are more limiting below about 2 w/o enrichment in both the homogeneous and heterogeneous systems.

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

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

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

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

14.3.1.2.2 Solid Angle Model

When criteria for the surface density model cannot be met for interacting subcritical units, the spacing may be established by the solid angle method described in Reference 5, subject to the limitations described therein. If this method is employed in a nuclear safety evaluation, each criterion and limitation shall be addressed in the documentation of the safety evaluation.

14.3.1.2.3 Transport and Monte Carlo Codes

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

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

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

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

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

14.3.2 Accident Conditions

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

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

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

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

4) interactions with SNM in transit;

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

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

7) adverse changes in interspersed moderation and reflector composition;

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

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

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

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

When multiple events are correlated or follow as a natural consequence, they are treated as a single event. The facility has been designed so that no single postulated credible event can result in a predicted critical condition. Process or environmental design changes or other engineered safeguards are given a higher priority over administrative controls in achieving this process design criterion.

14.3.3 Process Modelling

14.3.3.1 Computer Codes

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

For the first class of heterogeneous lattice calculations noted in the above paragraph, a four broad neutron group (three fast and one thermal) technique is needed. These few group cross sections are generated by the CEPAK lattice program. CEPAK is a synthesis of the FORM(6), THERMOS (7) and CINDER (8) codes. These codes are interlinked in a consistent way with inputs from differential cross section data from an 83 group library. Modifications have been applied to the U-238 resonance integral to correct for a recognized over-estimation of that quantity in ENDF/B-IV. The entire neutron spectrum is represented in 83 neutron groups between 0 and 10 MeV. Neutron leakage in a single Fourier mode is represented by either P-1 or B-1 approximations to transport theory throughout this entire range. Resonance shielding is determined analytically; the Hellstrand (9) correlation is employed for U-238, with appropriate adjustments guided by Monte Carlo calculations of resonance capture in U-238 so as to provide agreement with selected measurements of the conversion ratio. For clad UO2, appropriate Dancoff correction factors are determined for uniform lattices. For heterogeneous lattices, this calculation is extended to include the heterogeneities by nearest neighbor approximation. In some cases the GGC-3 Code (10) may be employed to calculate spectrum weighted few group cross sections for structural materials or trace element materials not in the CEPAK multigroup cross section library.

Few group spatial calculations for the heterogeneous lattices may be done with either the DOT II W transport code(11) or the KENO-IV Code (12). The DOT code is generally employed for the less complex geometric representations, whereas the KENO code can be employed for both simple and highly complex geometries.

In the second class of heterogeneous lattice calculations, i.e. low H/U lattices hydrogen calculations the KENO-IV code is employed for the determination of the effective multiplication factor. The NITAWL subroutine is employed to generate self-shielded 123 group cross sections from the 123 super group XSDRN library (14)-DLC-16. The resulting library is collapsed into a homogenized 16 broad group library in a typical heterogeneous lattice cell using the XSDRNPM Code (13). The latter code is also used to obtain 16 broad group cross sections for other regions containing structural and/or moderating regions. This code sequence is used for heterogeneous lattice calculations at low hydrogen density where approximations in the CEPAK lattice code are no longer valid.

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

14.3.3.2 Basic Cross Section Libraries

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

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

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

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

14.3.4 Administrative Control Models

14.3.4.1 Mass Controls

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

Alternative mass limit controls may be employed for certain operations which, in reality, are multiparameter controls. For example, a mass limit associated with specific containers (volume limit); in addition, controls on internal and/or external moderation limits may be imposed. Such administrative control models form the basis for the following limits specified in: 1) a 35 kg mass limit for UO2 in a closed 5 gallon, or less, UO2 container, and 2) a one foot spacing between subcritical units.

35 kg Homogeneous Mass Limit

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

12-Inch Separation

To assess the criticality safety of the 35 Kg mass limited, 5 gallon container in storage arrays employing a 12-inch separation between containers, a KENO analysis was run for an infinite planar array of such containers.

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

Separation (inches)	Multiplication Factor
12	0.9584 ± 0.0065
14	0.9414 ± 0.0081
16	0.9268 ± 0.0070

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

35 kg Heterogeneous Mass Limit

Here we consider a 35 Kg mass limit of heterogeneous material in containers having a volume of 5 gallons, or less. For a 5 gallon container the volume (19 liters) exceeds the safe volume of 17 liters for optimum moderation conditions at a UO2 enrichment of 5 w/o U-235 but is less than the critical reflected volume of 24 liters. Additionally, the mass of 35 Kg is just about equal to the minimum critical reflected mass of 35.4 Kg UO2 at the optimum rod diameter of 0.1 inches. This latter dimension corresponds to the optimum moderation condition for the mass limit curve at 5% of Figure 14-6.

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

An alternative method of looking at the criticality safety of UO2 pellets in a 5 gallon container is to determine the minimum critical mass as a function of a non-physical mixture of UO2 pellets and water homogeneously distributed over the volume of a 5 gallon container. Figure 14-13 shows such a relationship based on the data of Reference 3 for 0.1 inch diameter pellets. This Figure shows that a fully reflected 5 gallon container cannot go critical regardless of the mass of UO2 pellets within the container.

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

14.3.4.2 Internal Moderation Controls

The UO2 encountered in process operations at the Windsor Nuclear Manufacturing Facility is in the form of sintered pellets, pellet pieces, and recovered waste materials. The latter are handled under the SIU limits of Chapter 4. All completed fuel rods are stored in dry environments.

If a 5 gallon, or less, container of pellets or clean scrap is accidentally flooded, it may be safely stored in the concrete block storage area. This storage area was constructed to provide a controlled safe storage environment for such containers. Trays of pellets or clean scrap are stored or handled in a manner consistent with the 4 inch SIU slab limit of Table 4-2. In the fuel pellet storage racks, multiple levels of up to 4 inch thick slab arrays are employed. This storage device employs a stainless steel cover to preclude water from the interior of and between the slab configurations within the enclosure. Also see the discussion in Section 14.3.4.4.

Fuel rods when stored in the fuel rod storage boxes employ an administrative limit on height of the rod array in the box of 6 inches. This limit is facilitated by the box wall height of only 5 3/8 inches. Filled fuel rod storage boxes are stored in the fuel rod box storage area which is a stainless steel enclosed storage area to preclude flooding of these filled fuel rod storage boxes. When rod storage boxes are removed from the storage area, the number and configuration of the removed boxes are controlled by the handling equipment so as to assure criticality safety of the various configurations in the event of flooding of these containers. The principal basis for criticality safety of these arrays is the low water to fuel ratio achieved by the stacked fuel rod configurations. See the discussion in Section 14.3.4.5.

14.3.4.3 External Moderation Controls

External moderation controls have to do with the control of moderating material external to the container bearing the SNM. Firefighting techniques in the Windsor Fuel Manufacturing Facilities and the Product Development Laboratories are based primarily on the use of sprinkler and dry chemical systems; high pressure water hoses are not allowed in Building 17 but may be employed in the criticality areas of Building 5.

Care has been taken in the routing of water and steam pipes to avoid areas where large amounts of SNM are stored or processed. Thus, in the event of a rupture in these water or steam lines, there will be a reduced concern over potential criticality concerns. Analyses of the effective mist density resulting from the sprinkler system serve as a basis for definition of external moderation limits (see Section 14.7).

In most cases equipment has been designed to be safe in the event of complete reflection, e.g., a tight fitting reflector of water. There may be cases where such a precaution does not exist. In these instances a barrier is employed around the containers to prevent the approach of significant moderating material to within one foot of the wall of the container. This barrier modifies the criticality limits of the container such that it may be treated as a partially reflected container. A measure of the effect of an annular gap between a container of U(93) solution and a six-inch thick annular water reflector is given on page 19 of Reference 1. It is concluded that for a gap of 7 inches, the reflector savings is just one half that of a close fitting water reflector.

14.3.4.4 Slab Limit on Pellets

The SIU slab limit for heterogeneous UO2 at 5 w/o U235 is 3.5 inches (Figure 14-10). The purpose of the discussion to follow is to show that a 4 inch slab limit provides adequate criticality safety for randomly loaded pellets in trays or

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

Sintered UO2 pellets of 0.4 inch diameter when randomly loaded in a container pack to an average density of 5.95 g/cc with a one sigma variation of 0.264 g/cc, as determined by a series of 14 measurements.

Thus, at a 95% confidence level, the minimum average density is not less than 5.686 g/cc, which corresponds to a volume ratio of H2O to UO2 of 0.747. At this UO2 density, the minimum critical slab thickness is significantly greater than 4.17 inches for full reflection conditions. The data of Reference 3 does not extend much past the minimum slab thickness into this low H/U regime but the data of Reference 2 does. Figure 14-14 shows the data of Figure I.E.16 of Reference 2 for 5 w/o U-235 replotted on a log. log. grid to permit a more linear extrapolation to lower volume ratios. At a volume ratio of 0.747, the slab thickness is 7.2 inches. It should be noted that the data of Reference 3 does appear to indicate a smaller minimum critical slab thickness condition for pellet diameters of 0.4 inches by about 0.3 inches (Figure 14-6).

For conservatism, the minimum critical slab thickness deduced from Reference 2 is taken to be 7.2 inches and corrected downward by 0.3 inches to 6.9 inches. Dividing by a safety margin of 1.2 results in a slab limit of 5.7 inches. For trays containing pellets loaded in a random fashion, an administrative slab limit of 4.0 inches is adopted for UO2 pellets having enrichments of up to 5 w/o U-235. For trays containing pellets loaded in a non-random fashion, as with the corrugated separators, the SIU limits deduced from Figure 14-10 are employed, i.e., at optimum moderation and pellet diameter.

Although the data employed in the above evaluation of a safe slab limit was for an internally moderated and reflected slab of UO2 pellets, the four-inch slab limit on UO2 pellets has been examined by KENO analyses for purposes of gaining confidence in the safety of the deduced limit and to explore the sensitivity of the slab limit to external and internal moderation.

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

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

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

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

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

Conclusions drawn from these results are as follows.

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

2) For a range of spacings of up to six inches between the 4 inch high modules of the slab and for a range of water densities in the intra-module spaces varying between zero and full density, the effective multiplication factor is less than the value with zero spacing. Thus, the introduction of extraneous moderating materials between components of a large array of pellet trays arranged in a slab configuration will not result in an increase in the effective multiplication factor.

14.3.4.5 Array Size Limits on Fuel Rods

The purpose of this section id to explore the criticality limits on rod spacing and, in particular, the pitch or alignment of rods within a storage box to support the limits of Table 4-3. This evaluation demonstrates the safety of an open square pitch even in an infinite slab meeting the administrative 6 inch slab limit. This, in combination with the small width and height of the storage boxes, does not allow a sufficient moderation, even in the case of a random loading of rods within the box.

Completed fuel rods are subject to special handling instructions to avoid deformation of the clad tube. Consequently they are handled and stored in a limited number of configurations which provide the required mechanical support to the fuel rod when in a horizontal or near horizontal position. Most operations are analyzed explicitly to assess the criticality safety of the operation, e.g., the fuel rod transport cart, the pre-stacking operation, and the (covered) fuel rod storage area. The remaining operations involving general handling, transfer, and inspection operations do not usually involve large quantities of fuel rods, i.e., typically the equivalent of two fuel rod storage boxes or less. Consequently the types of arrays are few in number and can be generalized into three classes.

1) A slab array of rods on an inspection table or other flat surface involving inspection type operations,

2) Operations involving an individual fuel rod storage box, and

3) Operations involving two fuel storage boxes.

Slab arrays of fuel rods on flat inspection surfaces, because of the loose nature of the rods, generally assume a slab configuration consisting of a single, or at most, double layer of fuel rods. Thus the SIU heterogeneous slab limit of Chapter 4 provides a very conservative criticality limit for this configuration.

Fuel rod storage boxes are 8 inches wide, 5 3/8 inches deep, and approximately 14 feet in length. The use of an administrative limit of 6 inches on the height of the fuel rod array within a given fuel rod storage box provides adequate margin in the loading of the box since it exceeds the box wall height by five eighths of an inch or the equivalent of nearly two fuel rods in a hex array. A 6 x 8 inch rectangular array has the same buckling as a cylinder of diameter 7.97 inches. Since the latter diameter is less than the SIU heterogeneous limit for a cylinder, a single (isolated) storage box having fuel rods stacked to a 6 inch height is less reactive than the SIU limit because the latter is for the case of optimum moderation. In the case of a single box, the pitch of the rods is not limited.

The fuel rod storage box transport cart is employed to transport up to two (2) fuel rod storage boxes in a co-planar array approximately 30 inches above the floor. The two fuel rod boxes are positioned on individual roller beds such that the minimum spacing between the boxes is approximately 1.5 inches and the closest distance of approach between the outer edge of a fuel rod box and adjacent equipment is one foot.

A conservative approximation to the fuel rod storage box transport cart is to assume the two storage boxes are immediately adjacent to each other and are filled to the administrative limit of 6 inches with fuel rods. A buckling conversion of the 6 by 16 inch cross sectional area of the two boxes yields a 9.62 inch cylinder diameter. Since this exceeds the SIU heterogeneous cylinder limit by 15%, a consideration of physical limitations on the degree of moderation is necessary to show adequate criticality safety. Starting with the 9.62 inch cylinder diameter equivalent to the two storage boxes, 0.3 inches is added to cover the apparent bias between the 0.4 inch pellet data in References 2 and 3. Next, the safety margin factor of 1.1 is applied. The net result is that the inferred cylinder diameter is 10.91 inches. Referring to Figure 1.E.9 of Reference 2, it can be concluded that the critical volume ratio of water to UO2 is 1.35. At this volume ratio, 0.4 inch diameter unclad fuel rods are spaced 0.146 inches apart on a square pitch. Since it is not physically possible for this type of uniform lattice to exist in the storage box, even with misaligned rods, there are no practical limits on the alignment of the fuel rods in the two boxes.

Subcriticality requirements for an infinite slab of unclad rods are also examined using the same approach. Once again, the data of Reference 3 does not extend into the under moderated regime sufficiently far to permit extrapolation to the grams U-235/liter of interest. Therefore, the data of Reference 2 is employed for the slab limit on 0.4 inch diameter unclad UO2 rods (Figure 14-14). Since the administrative limit on rod height in the storage box is 6 inches and a safety factor of 1.2 plus a bias of 0.3 in. (see Section 14.3.4.4) are desirable, it remains to deduce the value of the volume ratio of water to UO2 at which a reflected slab thickness of 7.5 inches is just critical. The data of Figure 14-14 shows that at a H2O to UO2 volume ratio less than 0.7, a 7.5 inch thick slab of 0.4 inch diameter unclad fuel rods is subcritical. To achieve this volume ratio, the fuel rods are not in contact; they are separated by 0.06 inches on a square pitch. If the rods are in contact on a square or triangular pitch, the H2O to UO2 volume ratio is less than 0.3. If the pellets are in clad tubes, the critical dimension would increase. However, credit for this reactivity effect would again depend on how clad was treated in the modelling used to construct the data curves of Reference 2.

Based on the above discussion, it is concluded that use of the 6 inch administrative limit on the stacking height of fuel rods in the rod storage boxes provides an adequate margin on nuclear criticality safety for a co-planar array of fully reflected fuel rod storage boxes. The configuration of rods within the fuel rod storage boxes may be either triangular or square pitch, or a combination of both. As demonstrated in the above discussion, the rod spacing in a single box is arbitrary since it can be treated as an SIU. For two fuel rod storage boxes on the rod storage box transport cart, the average rod spacing limit is 0.14 inches on a square pitch. Since this is physically impossible, the alignment of rods within the box is not a criticality issue. Similarly, when viewed on the basis of an infinite slab array, the average spacing limit is 0.06 inches on a square pitch. Here again, it is physically impossible to achieve this average spacing. Consequently, the misalignment of occasional rods within a box is of no consequence. It should also be noted that the 6 inch slab cannot be flooded above the 5 3/8 inch box wall, therefore the above reasoning is conservative. It is further noted that once the filled storage boxes are placed in the fuel rod storage area, moderation of the rods is precluded by the enclosure. In the absence of water, rod spacing within the box is arbitrary and has no limiting spacing requirements.

14.4 Fixed Poisons

No credit for fixed poisons is employed under this license.

14.5 Structural Integrity Policy and Review Program

All storage racks, furnaces, containment, and processing equipment which provide nuclear safety limiting parameters shall be designed to assure against failure under normal and reasonable overload conditions and under conditions of shock or collision foreseeable in the plant area. 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 to fire and corrosion. The individual engineer responsible for the purchasing or design of the new equipment shall assure that these criteria are incorporated into the design of the equipment.

14.6 Analytical Models and their Validation

14.6.1 Heterogeneous UO2 - Water Configurations

14.6.1.1 Four Broad Neutron Group Model

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

For purposes of validating the analytical modelling of these heterogeneous UO2 - water lattices with or without parasitic absorbers, the critical separation experiments (15) carried out by Battelle Northwest Laboratories are analyzed.

These experiments are concerned with the critical separation distance between water moderated, subcritical clusters of fuel rods with different fixed neutron absorber types in the gap between fuel rod clusters. The experiments were carried out in a 1.8 m x 3 m x 2.1 m deep tank provided with features specifically designed and built for these experiments. The fuel rods had an active length of 914.4 mm and diameter of 11.176 mm. The fuel was clad by 6061 aluminum having an outer diameter of 12.70 mm and wall thickness of 0.762 mm. A fixed, square pin pitch of 20.32 mm was employed in each different size fuel rod cluster. Figure 14-15 shows a top and end view of the experimental configuration. Data on the experimental configurations analyzed are given in Tables 14-1 through 14-3.

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

Normally, for two dimensional representations, the transport Code DOT-IIW is used. Because of the short fuel length, the three dimensional Monte Carlo Code KENO IV is used with six axial levels. Batches of one hundred neutron histories are used with the first four discarded. Calculated multiplication factors are shown in Table 14-4. For economy, about 50 neutron batches were run for most cases, however, because of their greater use in fuel storage analyses, about 500 neutron batches were employed for the plain stainless steel and boral experiments.

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

14.6.1.2 Sixteen Broad Neutron Group Model

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

Critical experiments on the interposition of low hydrogen density materials between four 18 x 18 clusters of 4.742 w/o enriched UO2 rods were performed by the Department of Nuclear Safety of the French Atomic Energy Commission and reported in Reference 16. The fuel rods are spaced on a square pitch of 13.5 mm, contain UO2 pellets 0.790 cm in diameter, and are clad in aluminum tubes 0.94 cm O.D. with a wall thickness of 0.12 cm; the elements are 100 cm long. Figure 14-17 shows the experimental setup. The four fuel clusters are supported by a mobile device which allows them to move along orthogonal directions in a horizontal plane.

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

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

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

The computer codes employed in this analysis are KENO IV, NITAWL, and XSDRNPM. The reference microscopic cross section library is the 123 group super - XSDRN library, DLC-16 (14). The NITAWL and XSDRNPM Codes are used to generate 16 broad neutron energy group cross sections. NITAWL is used to generate self shielded 123 group cross sections from the 123 group super-XSDRN library (DLC-16). The resulting working library is then collapsed into a homogenized 16 energy group library in a typical fuel pin cell environment using XSDRNPM. XSDRNPM is also used to obtain separate 16 group cross section sets for structural materials and external moderators.

The KENO model employed a homogenized fuel pin representation in the interior of the fuel rod cluster. The cross shaped box, the outside moderator, tank wall, lattice grid, fuel pin lower plug, bottom plate and support plate are all explicitly represented. Table 14-5 summarizes the multiplication factors computed by KENO IV for 9 critical experiments.

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

Total Number of Results	7
Mean Value	1.00449
Standard Deviation	0.00643
95/95 Multiplier	3.34
95/95 Confidence Limits	0.022
Bias (u-1)	+0.00449

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

14.6.2 Homogeneous UO2 - Water Configurations

Validation of the KENO-IV code and Hansen-Roach cross sections, as distributed under the SCALE code system (12), is described in Reference 17. To ascertain whether the conclusions of the latter reference are applicable to homogeneous analyses for this license, the following comparisons were made.

1) The Hansen and Roach cross sections library was verified as being identical to that distributed under SCALE, and

2) Eight of the sample problems distributed with the code were run for purposes of comparing the calculated eigenvalues with those obtained by ORNL.

Table 14-6 summarizes the eigenvalues obtained by C-E and ORNL for each sample problem. The eigenvalues agree within the stated statistical deviation. Thus, it may be concluded that the conclusions of Reference 17 concerning bias and deviation are applicable to homogeneous analyses performed by C-E.

14.7 Special Controls

14.7.1 Moderation Control

Moderation control is implemented in the SNM handling areas of Building 17 by control of the sources of water. Water pipes are routed in a fashion so as to minimize the likelihood of water spraying on SNM in the event of a pipe rupture. Pipe sizes are such that, in the event of a pipe rupture, the water density is sufficiently low that it does not exceed that of the fire sprinkler system over a sufficiently large area so as to cause a criticality concern. Furthermore, storage areas for pellets and fuel rods are designed to preclude water from the stored material by metal enclosures. On this basis it is concluded that the mist density attributed to the fire sprinkler system bounds the moderation condition of interest for criticality safety evaluations.

Subsequent parts of this section provide an evaluation of the mist density attributable to the fire sprinkler system and the water film thickness on a fuel assembly.

14.7.2 An Estimation of the Water Volume Fraction Provided in the Assembly Room of Building No. 17

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 are evaluated separately.

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.

14.7.2.1 Scope

The estimation was performed exclusively for the sprinkler systems and room configuration shown in Figure 14-18.

The sprinkler system in Figure 14-18 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 14-18. These regions are delineated by the walls and dashed lines shown in the figure.

The estimation was based on the following assumptions:

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 (Reference 18).

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

14.7.2.2 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 14-19). 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

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

The water pressure drop to friction loss from the top of the riser to the region of interest is obtained from the Factory Mutual Pipe Friction Loss Tables (Reference 19).

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

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

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:

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

14.7.2.3 Calculations

The estimation of water volume fraction in air was performed separately from Regions A, B, and C shown in Figure 14-18. The following calculation procedures for each region are identified by numbers in accordance with those of Section 14.2.7.2.

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 14-18) 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 14-18.

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 (Reference 20). 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 (Reference 23). The time needed for 0.7 mm drops to fall from the sprinkler to the floor

$$= 27 \text{ ft}/11.5 \text{ ft/s}$$
$$= 2.35 \text{ s.}$$

Therefore,

$$\text{The water volume fraction} = \frac{(446)(0.039)(0.13368)}{(27)\ (30)\ (39)} \times 100$$

$$= 0.0074\%$$

<u>Region B</u>

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 14-18) of a 3-in pipe

   = 0.238 x 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

   = $0.86\ (30/61)^{1/3}$ mm
   = 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 to fall from the sprinkler to the floor

$= 27/11.5$ s
$= 2.35$ s.

$$\text{The water volume fraction} = \frac{(467)(0.039)(0.13368)}{(27)\ (30)\ (39)} \times 100$$
$$= 0.0075\%$$

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 due 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
$= 102.0$ 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 to fall from the sprinkler to the floor

$= 27/11.5$ s
$= 2.35$ s.

$$\text{The water volume fraction} = \frac{(992)(0.039)(0.13368)}{(27)\ (64)\ (39)} \times 100$$

$$= 0.0075\%$$

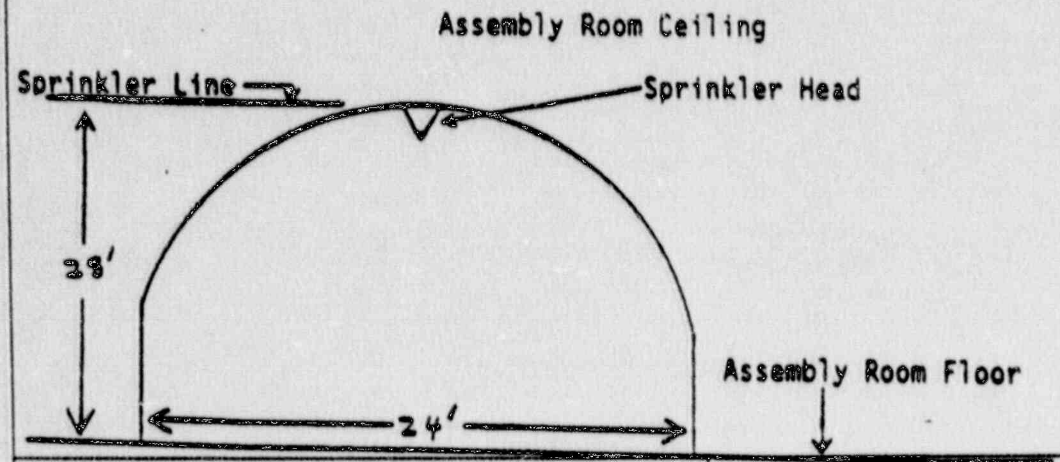
14.7.2.4 <u>Mist Density Calculation for a Single Sprinkler Head</u>

Density from a single sprinkler head anywhere in region A, B, or C, Figure 14-18.

1. 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.
= 8 ft^3/min.



2. Volume of Sprinkler Flow Paraboloid = V = 1/2 R^2H

Where = A constant = 3.14
R = Radius of spray of floor = 12'
H = Distance from sprinkler head to floor = 28 ft.
V = 1/2 (3.14) $(12)^2$ (28)
V = 6330 ft^3

3. Water Drop Size from 1/2" Sprinkler Head (Supplied by FMIC)

$$D_2 = \left(\frac{P_1}{P_2}\right)^{1/3} D_1$$

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 (Drop Size @ 100 psi) = $\left(\frac{30}{100}\right)^{1/3}$ (0.86)

D_2 = (0.89 (0.86) = 0.77 mm @ 100 psi

4. Drop Velocity

Reference drop velocity for 1 mm drop = 13 ft/sec
Drop Velocity @ 0.77 mm = 0.77 x 13 10 ft/sec

5. Time for Drop to Reach Floor

T = Drop time from 28 ft - $\frac{28ft}{10ft/sec}$ = 2.8 sec.

6. 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

14.7.2.5 Summary

The water volume fractions in Regions A, B, and C in the Assembly Room of Building 17 (see Figure 14-18) were estimated separately based on the sprinkler system and room configuration illustrated in Figure 14-18, and the water supply test data shown in Figure 14-19. The estimated water volume fractions in air in the above three regions are about 0.0075%.

14.7.3 An Estimation of the Water Film Thickness on Fuel Rods (In Fuel Bundles) During a Release of Water From the Sprinkler System

14.7.3.1 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.

14.7.3.2 Basic Information

1. Fuel Arrangement (Geometry)

   Fuel O.D. = 0.382 inches
   Fuel Pitch = 0.506 inches

2. Flow Rate

   For Region B of the storage room = 467 gal/min.
   (See Figure 14-19)
   Area of storage room = 30 x 40 = 1200 ft^2

3. Physical Properties

   at 15.7 psia and 77°F
   Water Density = 62.3 lb/ft^3
   Water Viscosity $s = 2.0 \times 10^{-5}\ lb_F\text{-sec}/ft^2$

at 14.7 psia and 50°F
Water Density = 62.3 lb/ft^3
Water Viscosity s = 2.73 x 10^{-5} lb$_F$-sec/ft^2

5. Area of a Single Fuel Lattice

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

6. Clad Perimeter of a Single Fuel Lattice

$$= \frac{(0.382\ )}{12} = 0.1\ ft$$

7. Water Flow Rate Per Fuel Lattice

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

8. Formula for Film Thickness

For this calculation references 21 and 22 are used.

$$= \left(\frac{3s\ g_c}{{}^2 g}\right)^{1/3}$$

where = film thickness (ft)
s = viscosity (lb$_F$-sec/ft^2)
= mass flow rate per unit width of wall (lb/ft sec)
= density (lb/ft^3)
g = acceleration by gravity (ft/sec^2)
g_c = conversion factor

9. Film Thickness Calculation

at 14.7 psia and 77°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 $= \left[\dfrac{3 \times 2.0 \times 10^{-5} \frac{\text{lb-sec}}{\text{ft}^2} \times 9.6 \times 10^{-4} \frac{\text{lb}}{\text{ft-sec}} \times 32.2 \frac{\text{ft}}{\text{sec}^2}}{(62.3)^2 \frac{\text{lb}}{\text{ft}^3} \times 32.2 \frac{\text{ft}}{\text{sec}^2}} \right]^{1/3}$

$= 0.00025$ ft $= 0.00295$ inches $= 0.0075$ cm

at 14.7 psia and 50°F

$$= \frac{1.54 \times 10^{-6} \frac{\text{ft}^3}{\text{sec}} \times 62.3 \frac{\text{lb}}{\text{ft}^3}}{0.1 \text{ ft}} = 9.6 \times 10^{-4} \frac{\text{lb}}{\text{ft-sec}}$$

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

$= 0.00027$ ft $= 0.0033$ inches $= 0.0083$ cm

14.7.3.3 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. 23). Dukler shows that his approach gives similar film thicknesses as the Nusselt approach (Ref. 21) at zero shear stress at the film/air interface and low Reynolds numbers (less than 300).

For the present case, the Reynolds number is:

$$Re = \frac{4}{s}$$

$$Re = \frac{4 \times 9.6 \times 10^{-4} \frac{\text{lb}}{\text{ft-sec}}}{2.0 \times 10^{-5} \frac{\text{lb-sec}}{\text{ft}^2} \times 32.2 \frac{\text{ft}}{\text{sec}^2}} = 5.9$$

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

14.8 Data Sources

1. H. C. Paxton, "Criticality Control in Operations with Fossil Material", LA-3366 (Rev.), Los Alamos Scientific Laboratory, 1972.

2. J. H. Chalmers, et al, "Handbook of Criticality Data, Volume 1, UKAEA AHSB(S) Handbook 1, 1965.

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Table 14-1

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 WT% U-235
ENRICHED UO2 RODS IN WATER

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

(1) Fuel rods on 20.32 mm square pitch.

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

(3) Rerun of Experiment 005

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

Table 14-2

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

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

(1) Error limits shown are one standard deviation.

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

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

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

Table 14-3

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 wt% U-235 ENRICHED UO2 RODS IN WATER WITH BORAL PLATES BETWEEN FUEL CLUSTERS (1)

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

(1) Error limits shown are on standard deviation.

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

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

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

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

Table 14-4

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 100 w/o Boron	1.00418	0.00273
26	304 S Steel 0.0 w/o Boron	0.99811	0.00279
34	304 S Steel 0.0 w/o Boron	0.00793	0.00297
35	304 S Steel 0.0 w/o Boron	1.00436	0.00290
32	304 S Steel 1.05 w/o Boron	0.99970	0.00524
33	304 S Steel 1.05 w/o Boron	1.01173	0.00491
38	304 S Steel 1.62 w/o Boron	1.00289	0.00512
39	304 S Steel 1.62 w/o Boron	1.00208	0.00506
20	BORAL	0.99585	0.00301
16	BORAL	1.00020	0.00288
17	BORAL	0.99519	0.00286
	Mean k_{eff} Value	1.00157	
	Standard Deviation	0.00419	

Table 14-5

KENO IV RESULTS FOR NOTED GAP WIDTHS

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

Table 14-6

COMPARISON OF KENO-IV CALCULATED EIGENVALUES FOR SAMPLE PROBLEMS

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

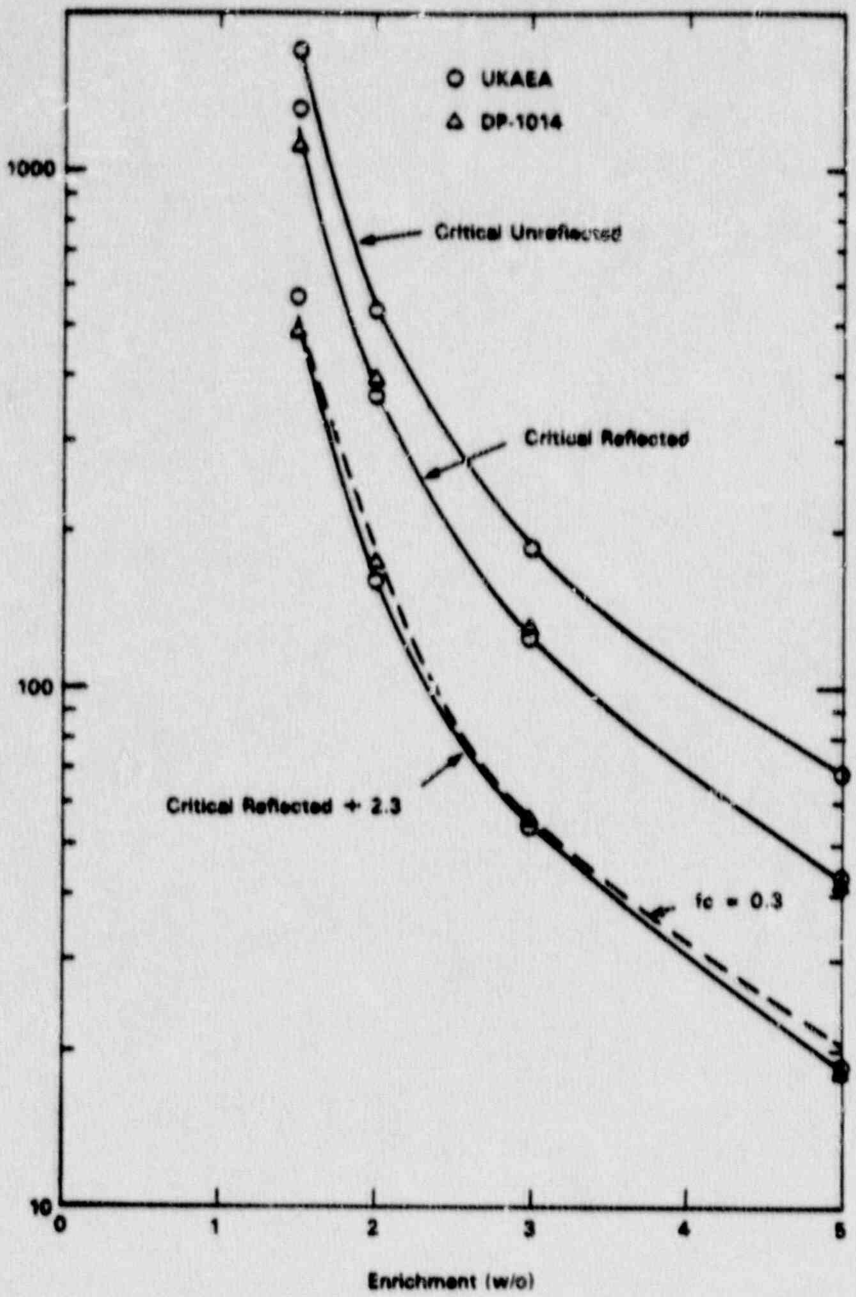

Figure 14-1
Minimum Critical Mass For Homogeneous UO2 Powder - H2O

PART II
SAFETY DEMONSTRATION

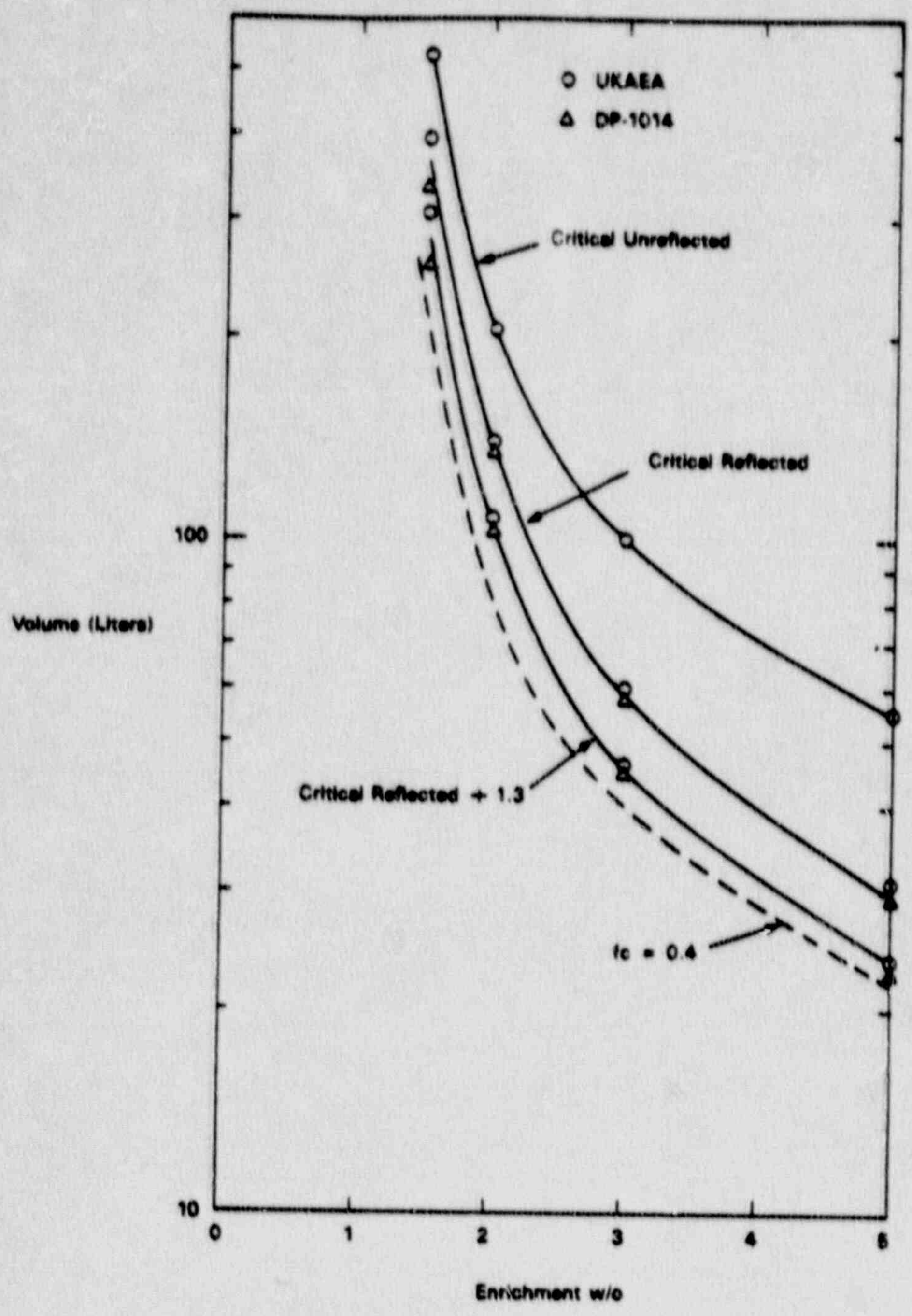

Figure 14-2

Minimum Critical Volume For Homogeneous UO2 Powder - H2O

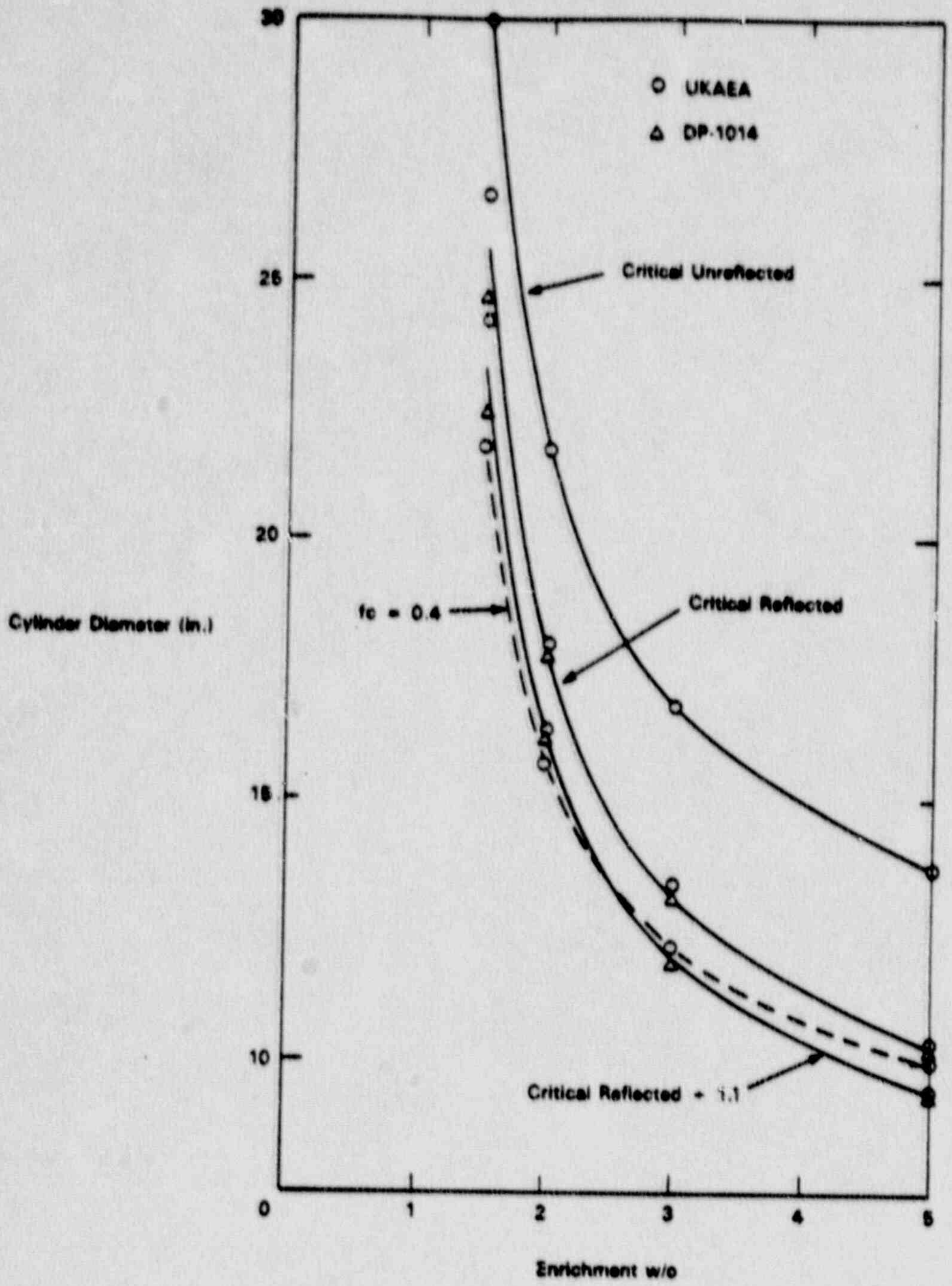

Figure 14-3
Minimum Critical Infinite Cylinder Diameter For Homogeneous
UO2 Powder - H2O

PART II
SAFETY DEMONSTRATION

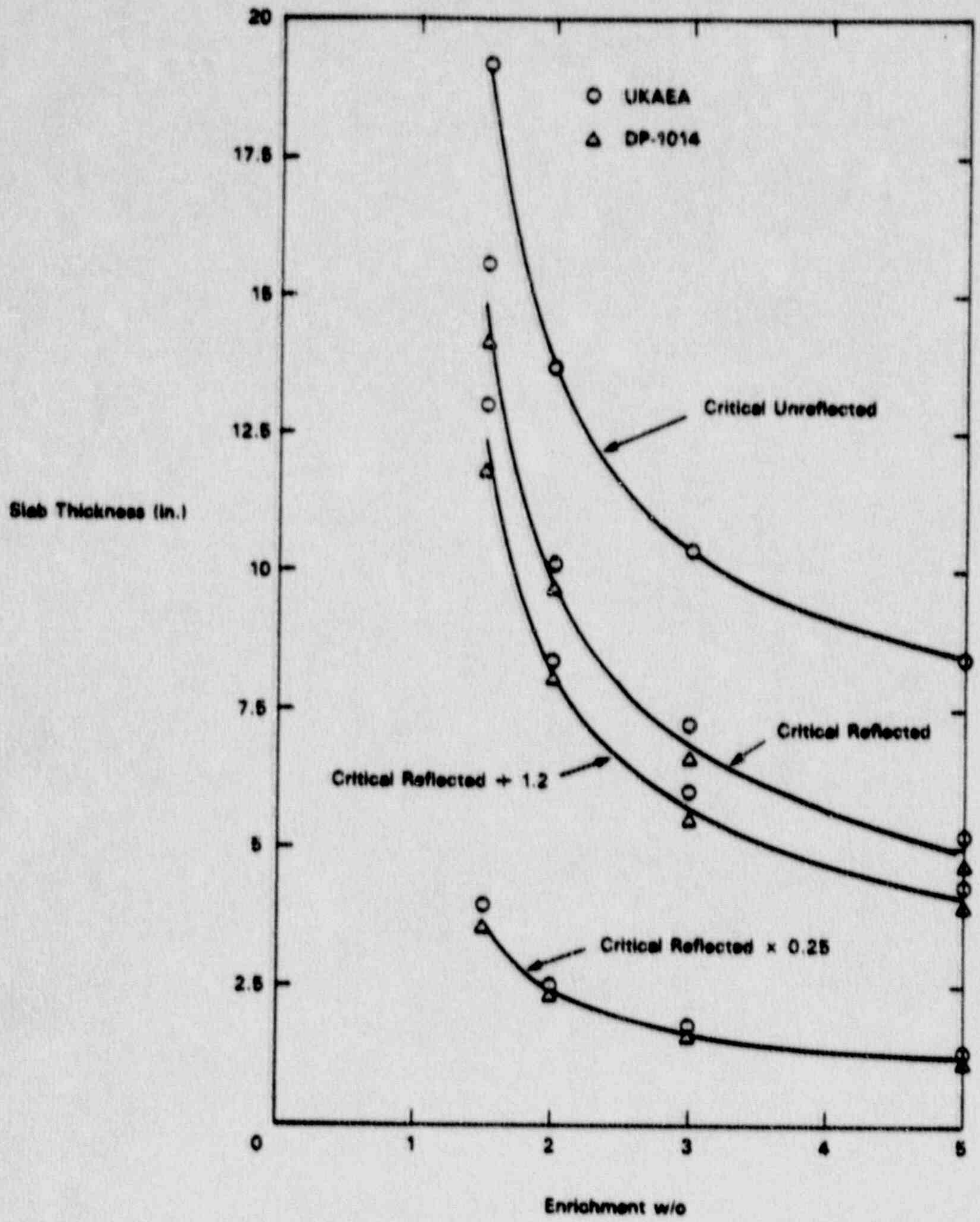

Figure 14-4
Minimum Critical Semi-Infinite Slab Thickness For Homogeneous Infinite Slab Of UO2 - H2O

PART II
SAFETY DEMONSTRATION

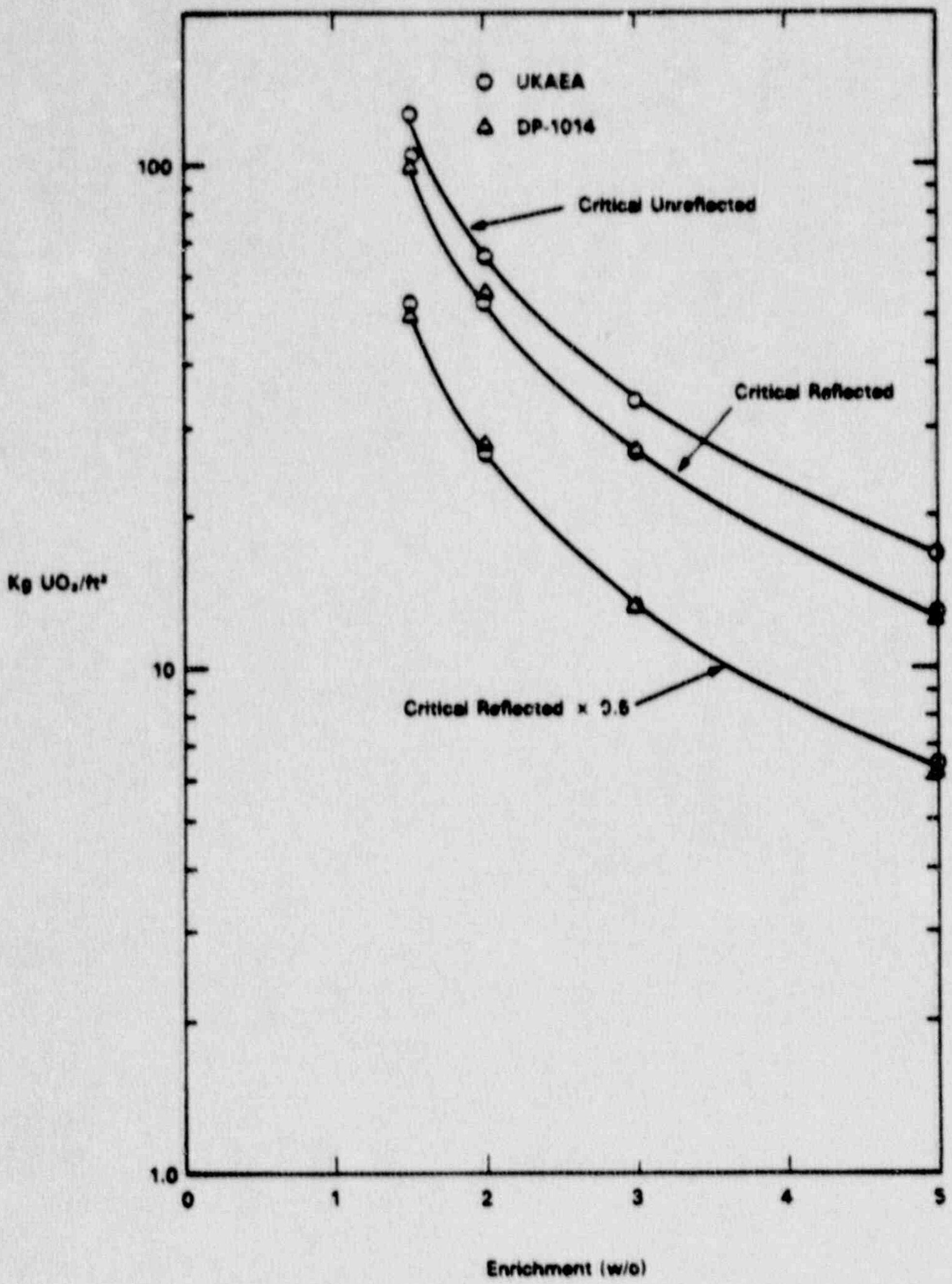

Figure 14-5
Minimum Critical Semi-Infinite Slab Surface Density
For Homogeneous U20 - H20

PART II
SAFETY DEMONSTRATION

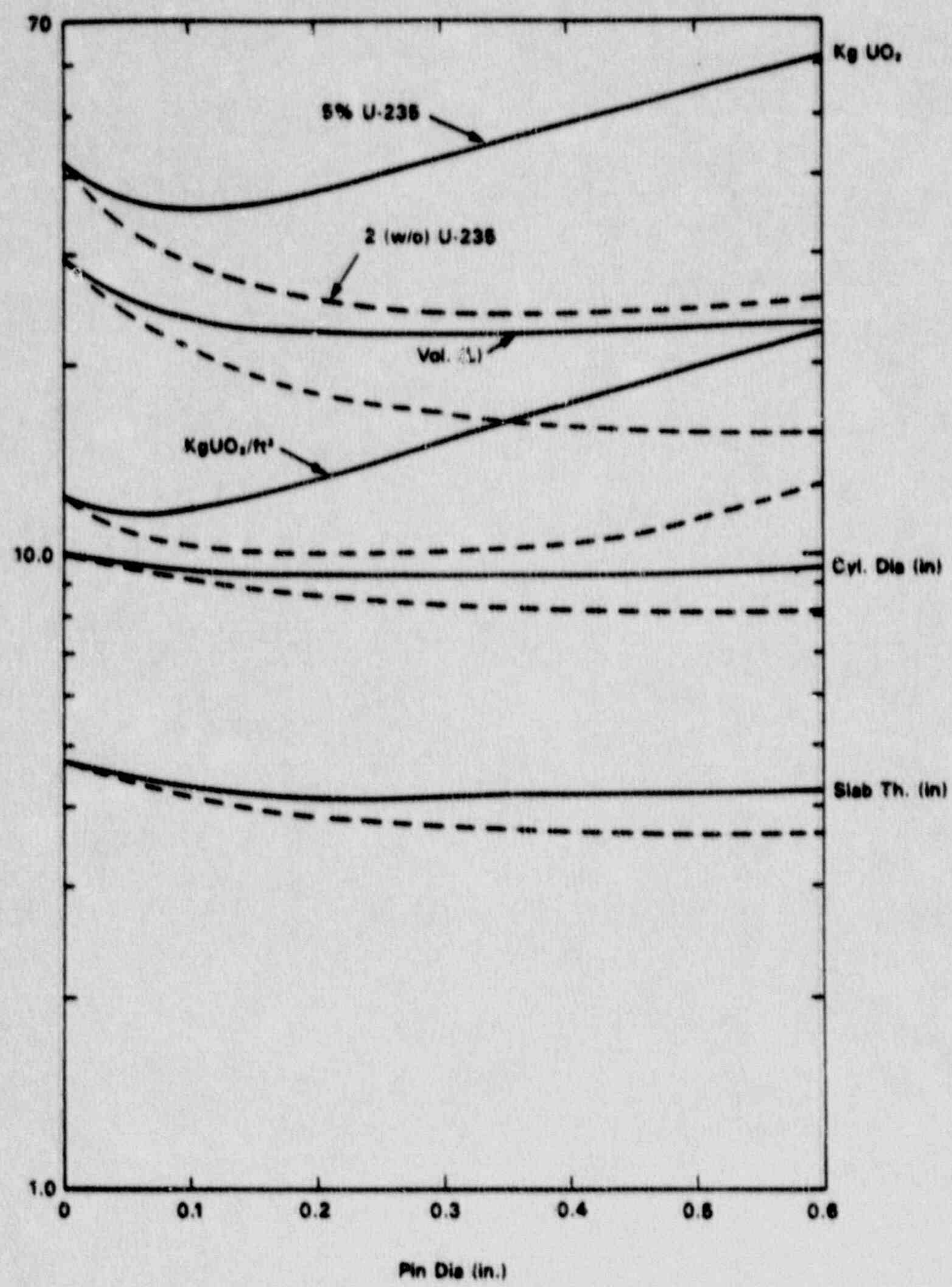

Figure 14-6
Parameter Dependence On Pellet Diameter (Data From DP-1014)

PART II
SAFETY DEMONSTRATION

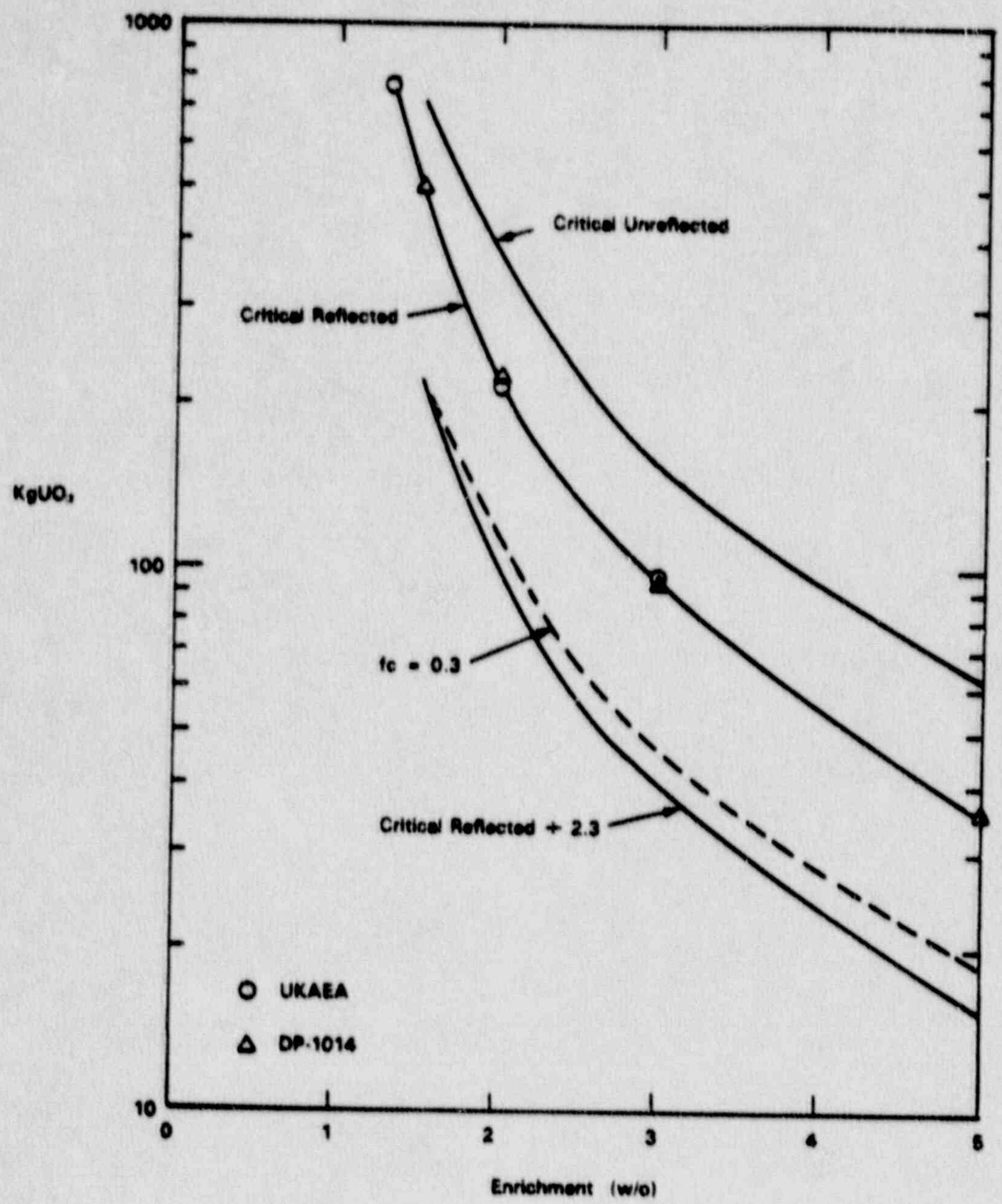

Figure 14-7

Minimum Critical Mass For Heterogeneous UO2 - H2O

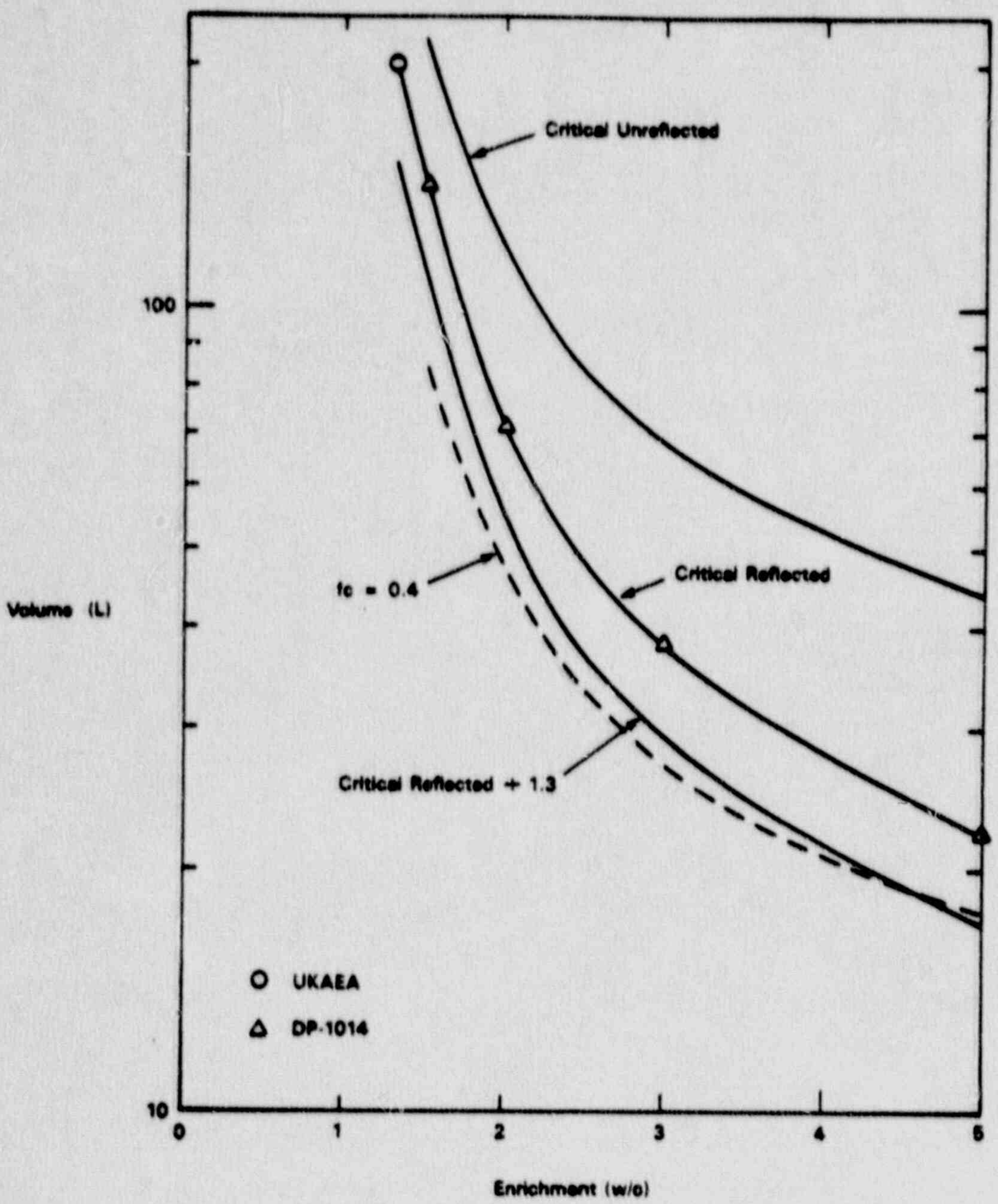

Figure 14-8

Minimum Critical Volume For Heterogeneous UO2 - H2O

PART II
SAFETY DEMONSTRATION

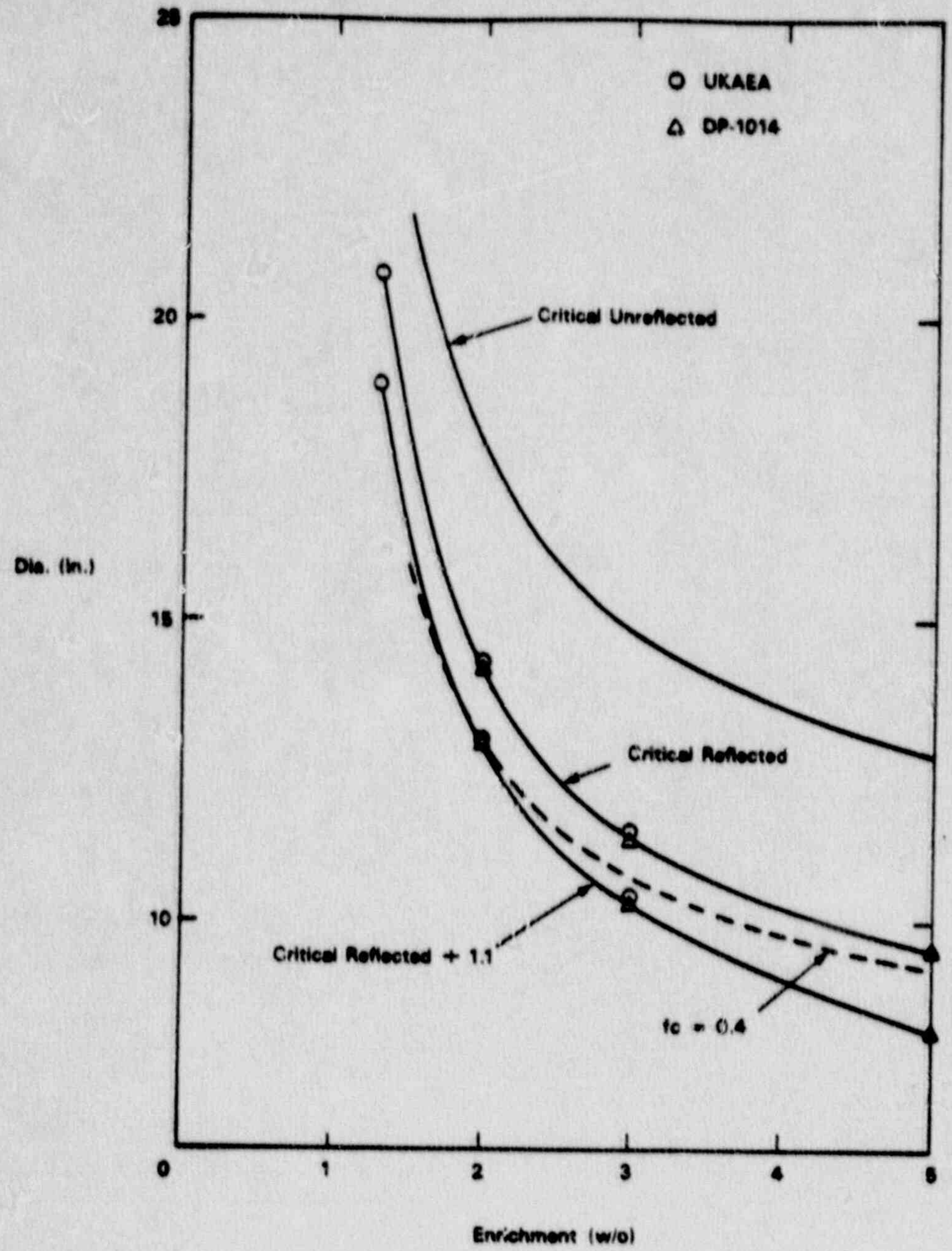

Figure 14-9

Minimum Critical Infinite Cylinder Diameter For Heterogeneous UO_2 - H_2O

PART II
SAFETY DEMONSTRATION

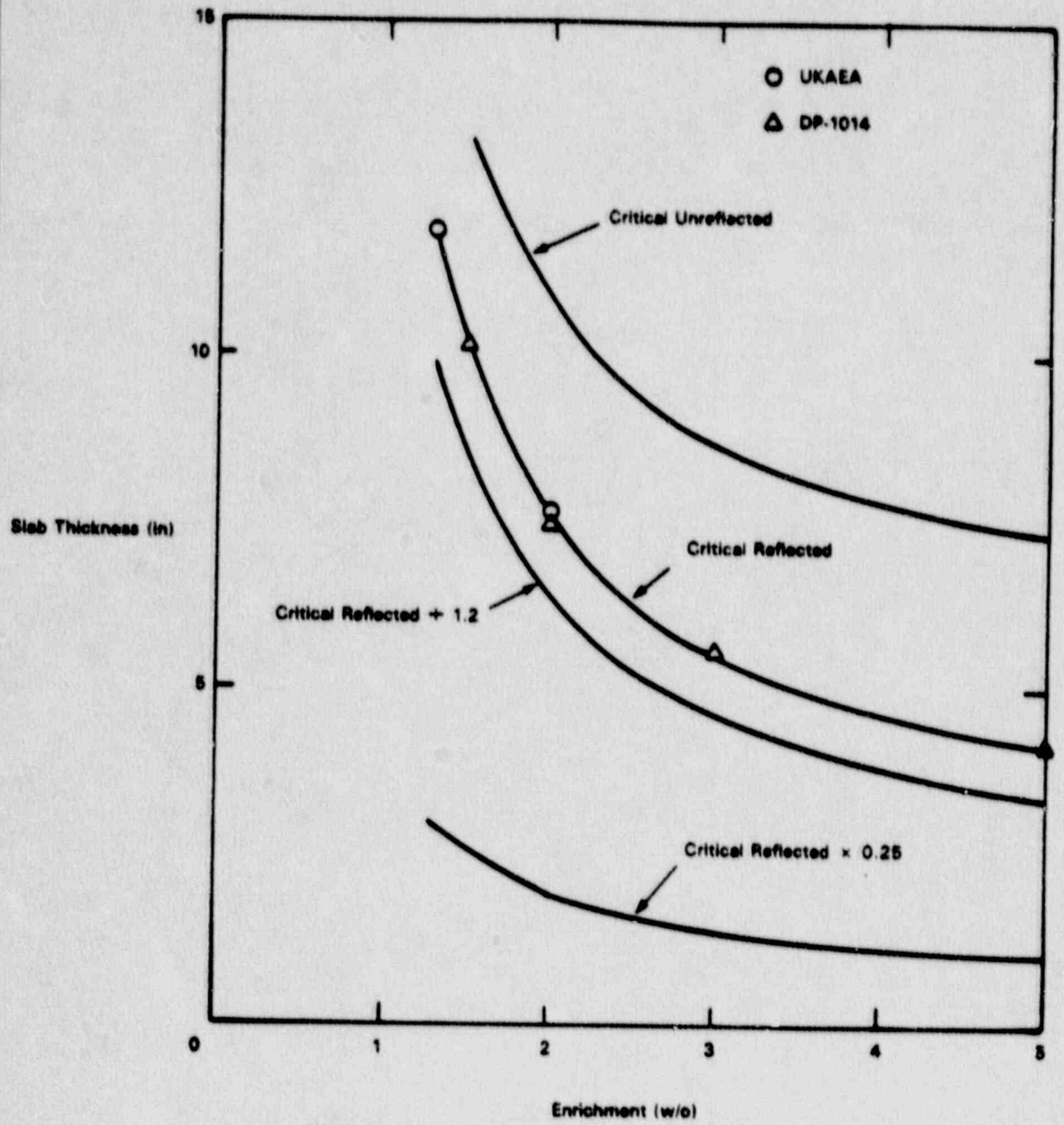

Figure 14-10
Minimum Critical Semi- Infinite Slab Thickness For Heterogeneous UO2 - H2O

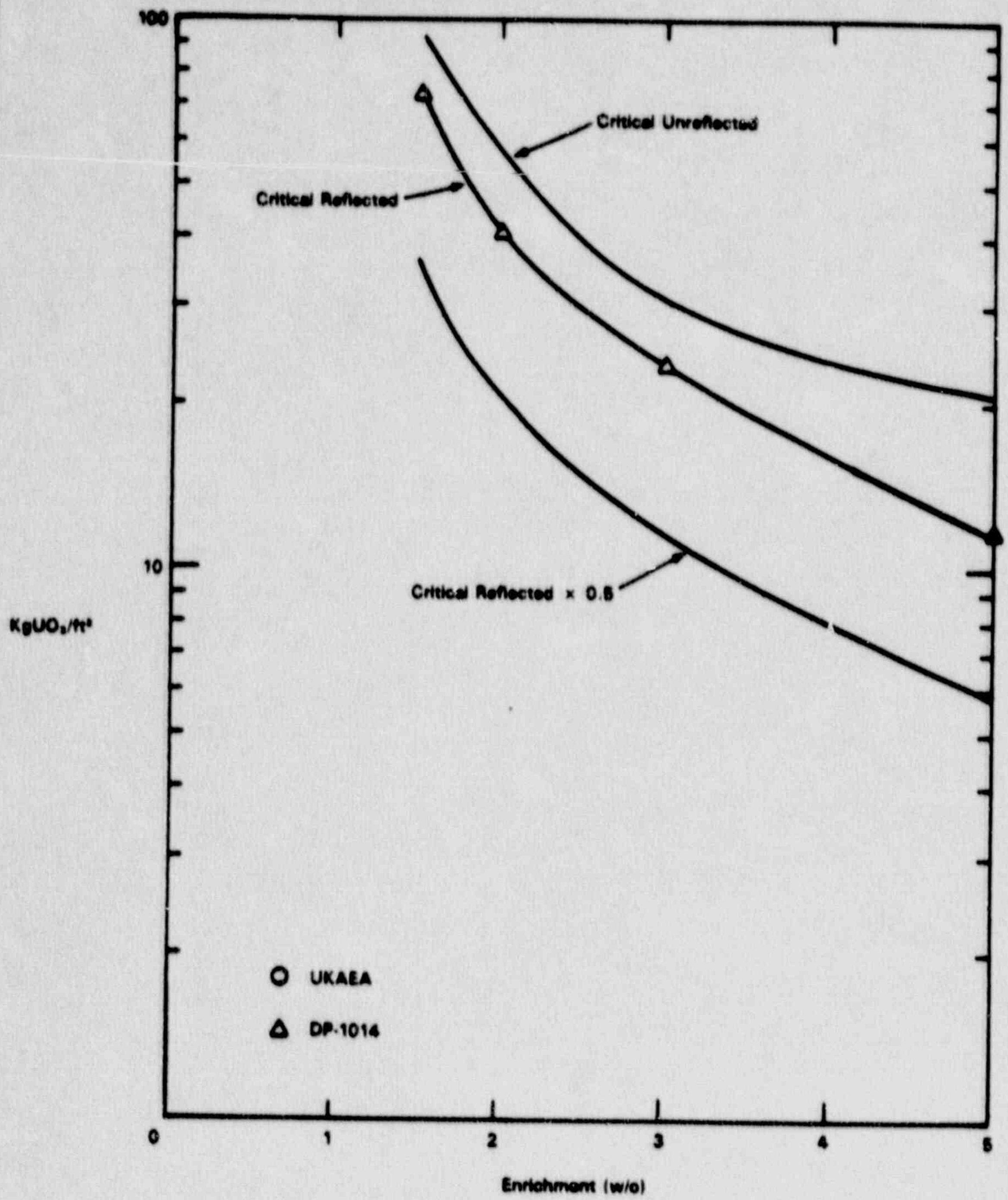

Figure 14-11

Minimum Semi- Infinite Slab Surface Density For
Heterogeneous U20 - H20

PART II
SAFETY DEMONSTRATION

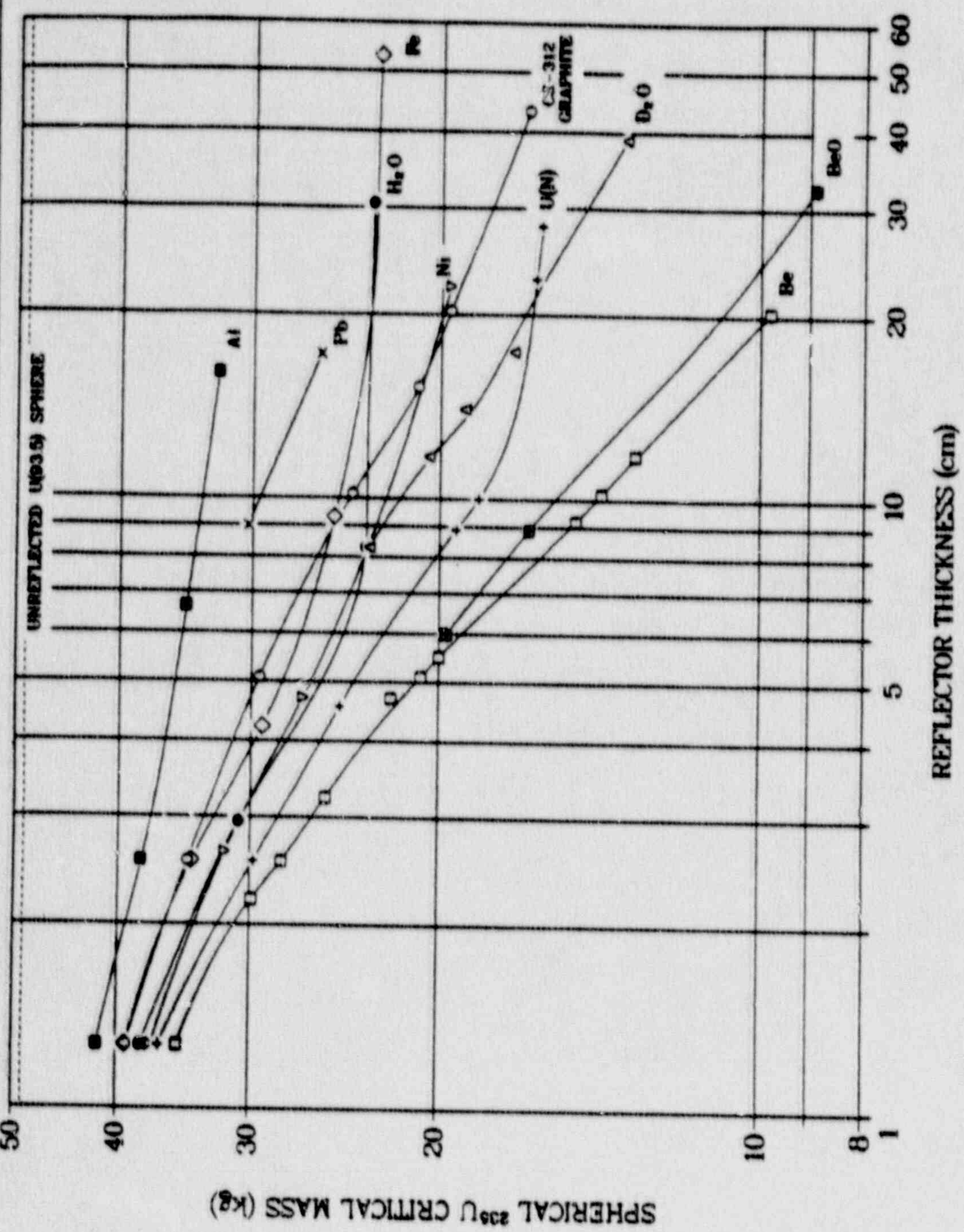

Figure 14-12

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

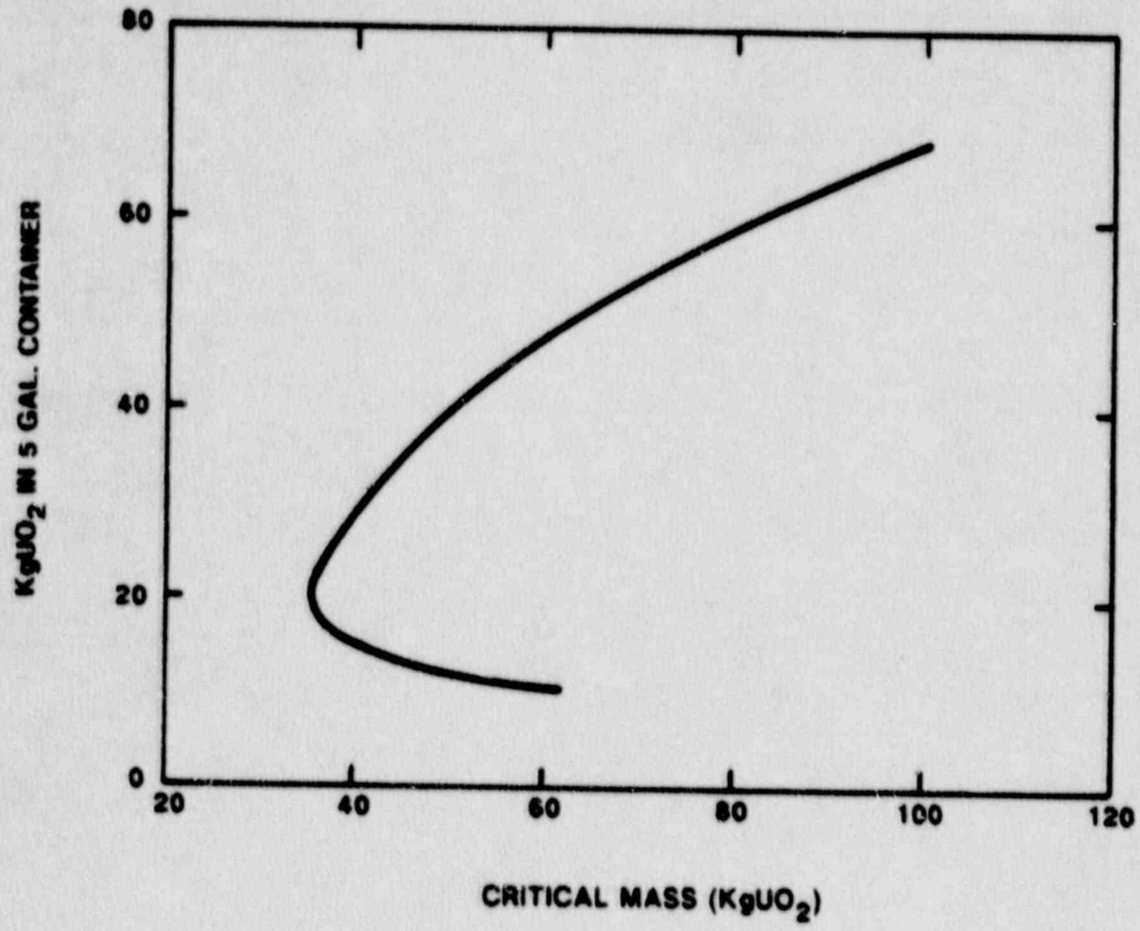

FIGURE 14-13

$KgUO_2$ (0.10" PELLETS) HOMOGEOUSLY DISTRIBUTED IN H_2O
IN 5 GAL. CONTAINER VERSUS CRITICAL MASS (5 W/O U-235, REF. 3)

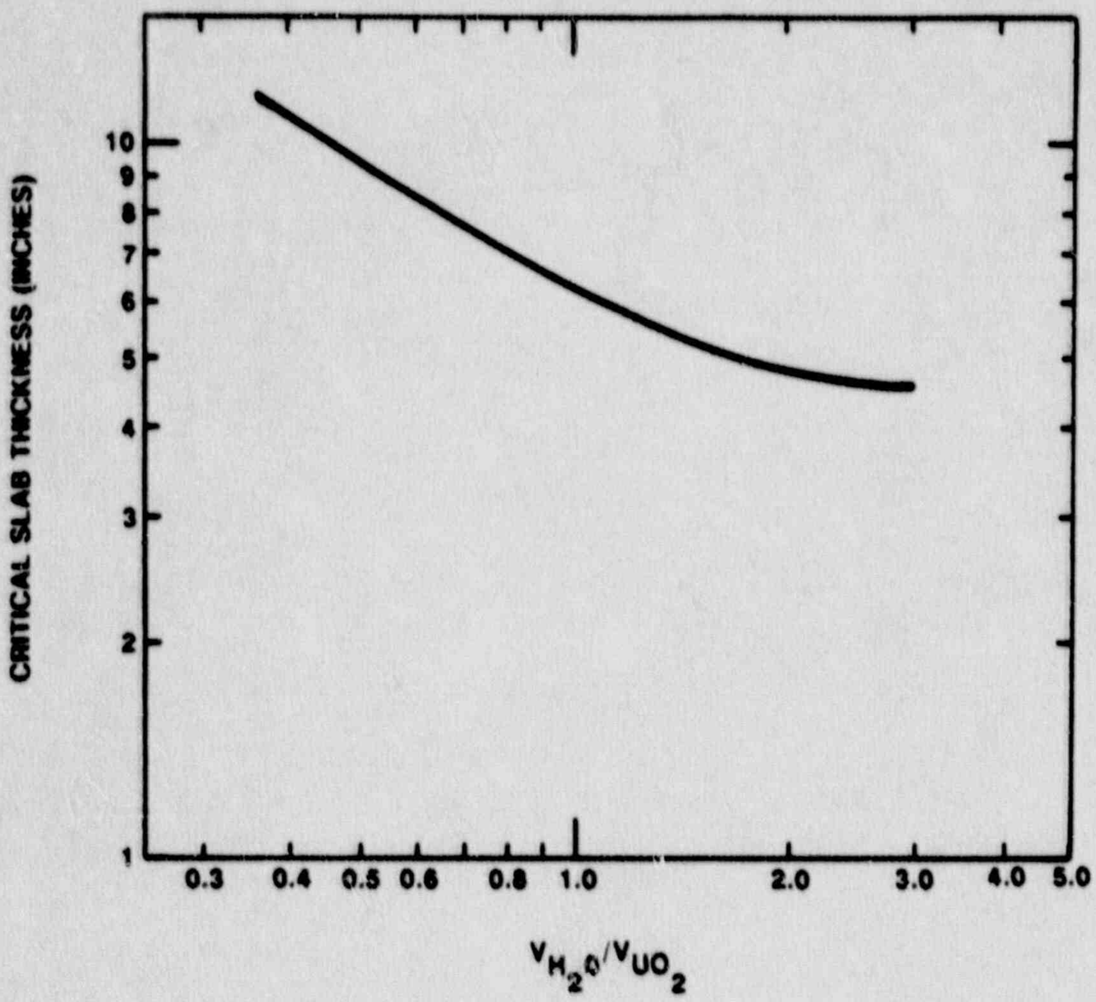

FIGURE 14-14

CRITICAL SLAB THICKNESS (IN.) VERSUS VOLUME RATIO OF H_2O TO UO_2 FOR 5 W/O U-235 (REFERENCE 2; 0.4 INCH PELLETS)

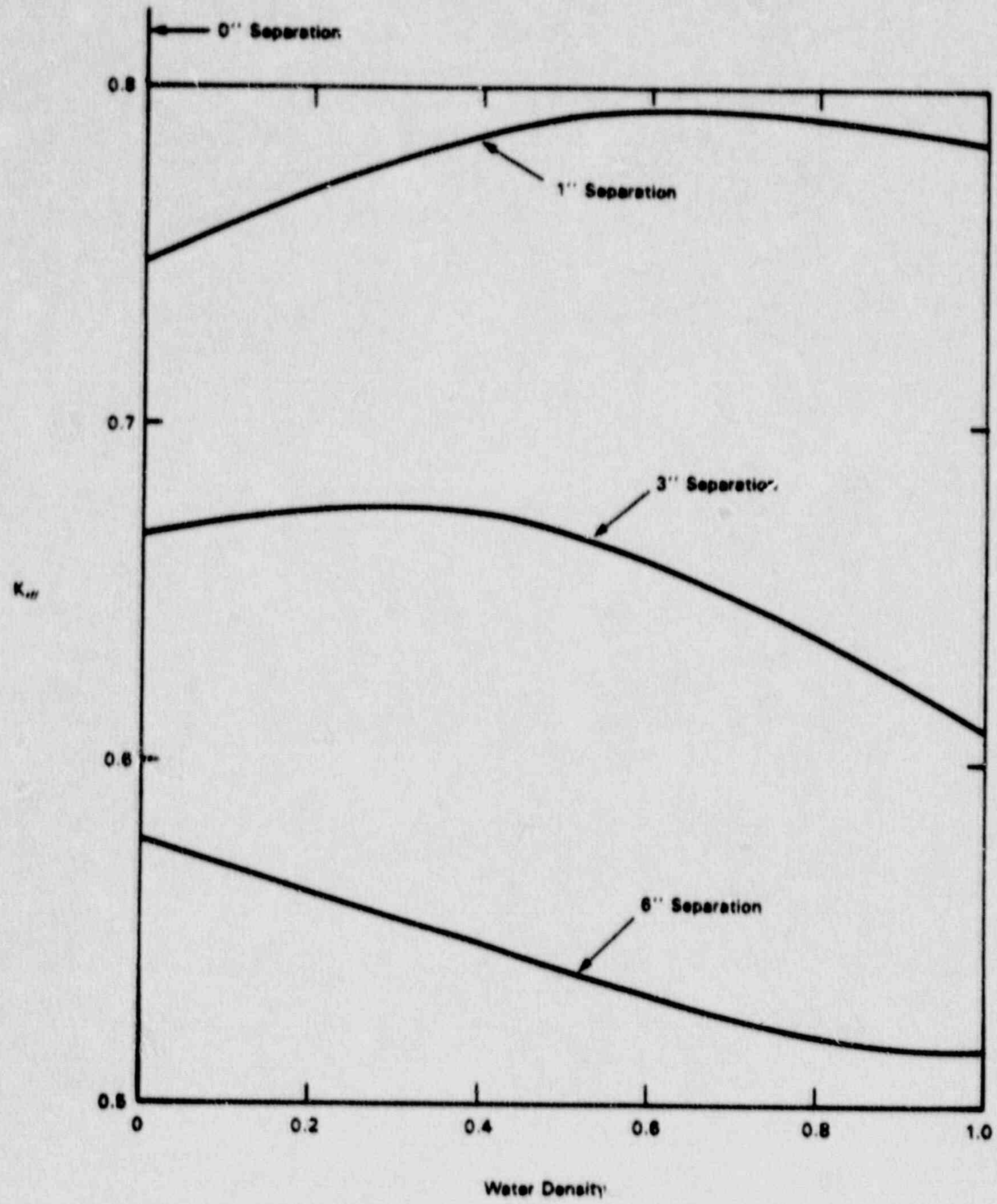

Figure 14-15

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

PART II
SAFETY DEMONSTRATION

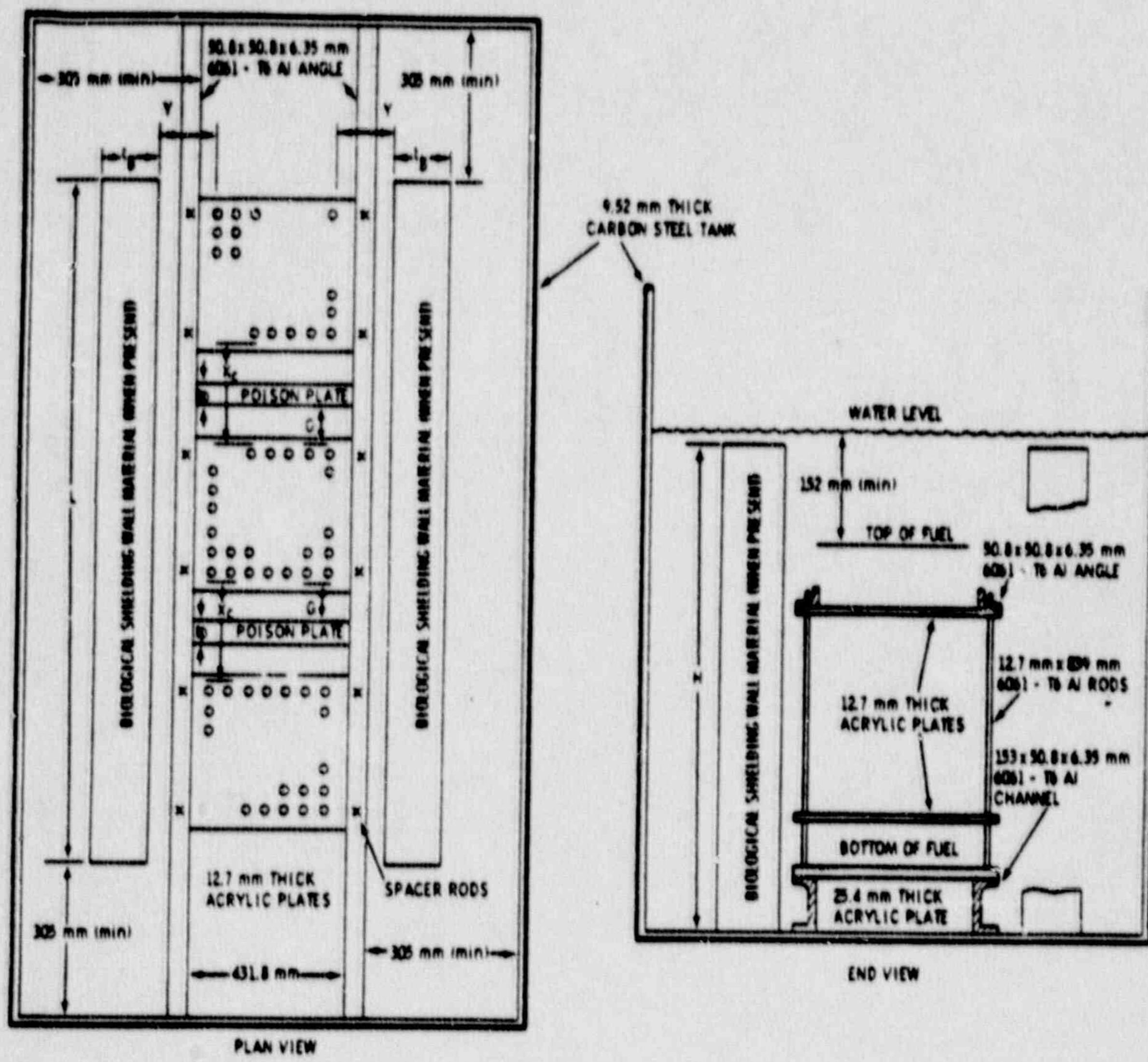

Figure 14-16

GRAPHICAL ARRANGEMENT OF SIMULATED SHIPPING CASK CRITICAL EXPERIMENTS

PART II
SAFETY DEMONSTRATION

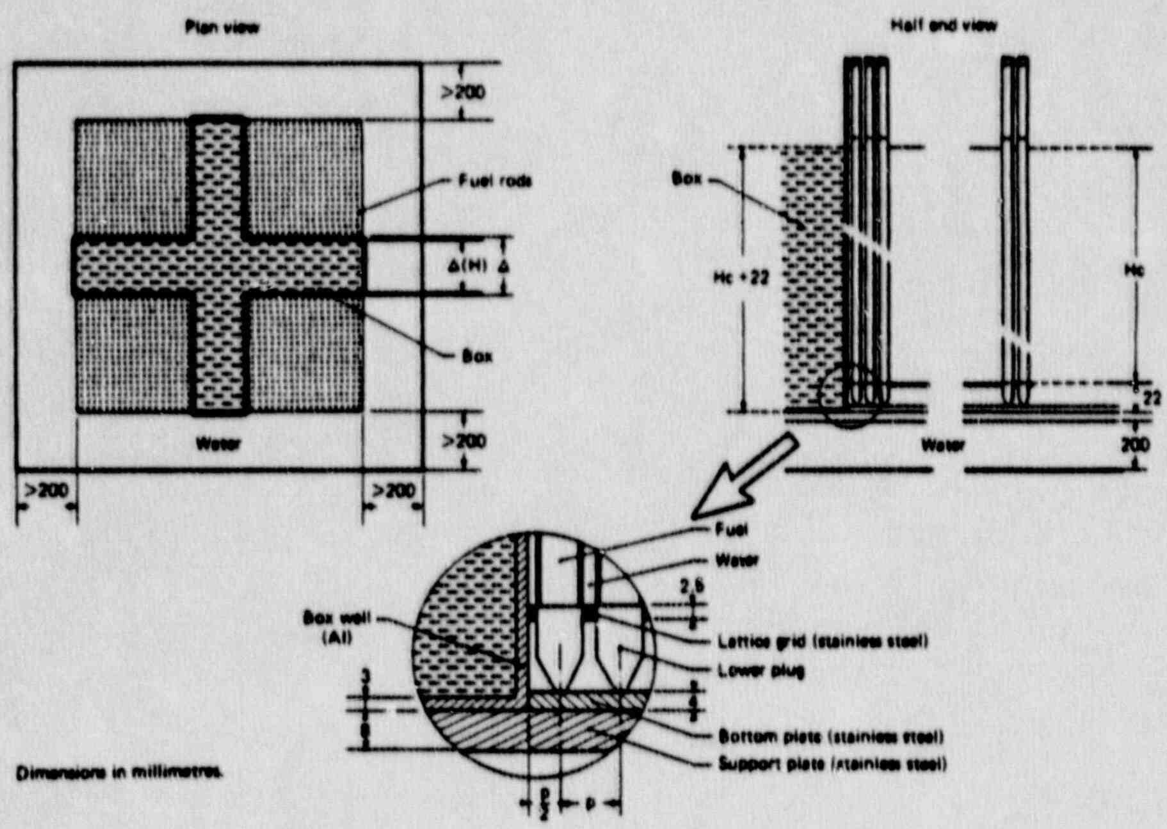

Figure 14-17

EXPERIMENTAL SETUP FOR EXPERIMENTS OF REFERENCE 18

PART II
SAFETY DEMONSTRATION

5. — - - — SIGNIFIES SEPARATION ZONE BETWEEN SPRINKLER LINES.

4. - - - - - - SIGNIFIES SPRINKLER DISTRIBUTION LINE.

3. ——— SIGNIFIES 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 FEET.

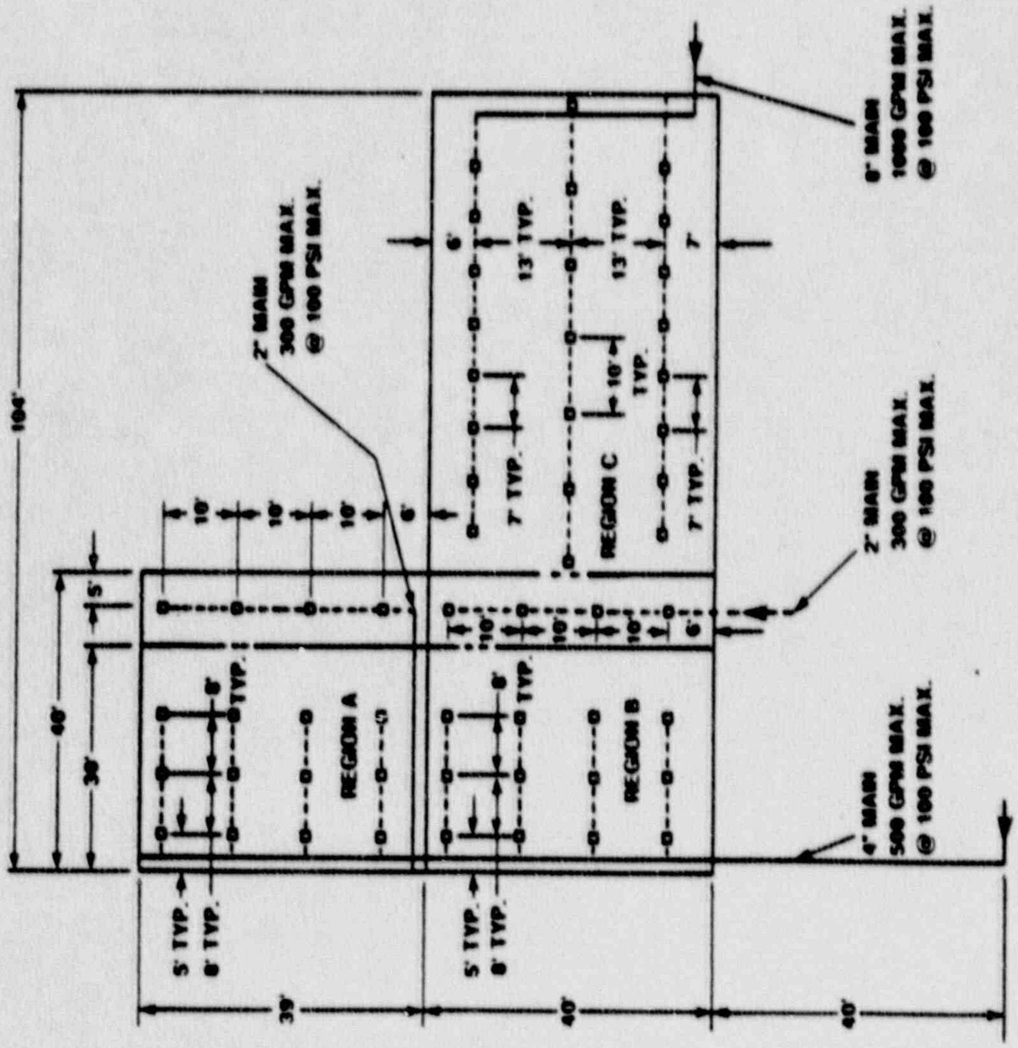

Figure 14-18

LAYOUT OF SPRINKLER SYSTEM IN PELLET SHOP ANNEX AND BUNDLE ASSEMBLY ROOM

PART II
SAFETY DEMONSTRATION

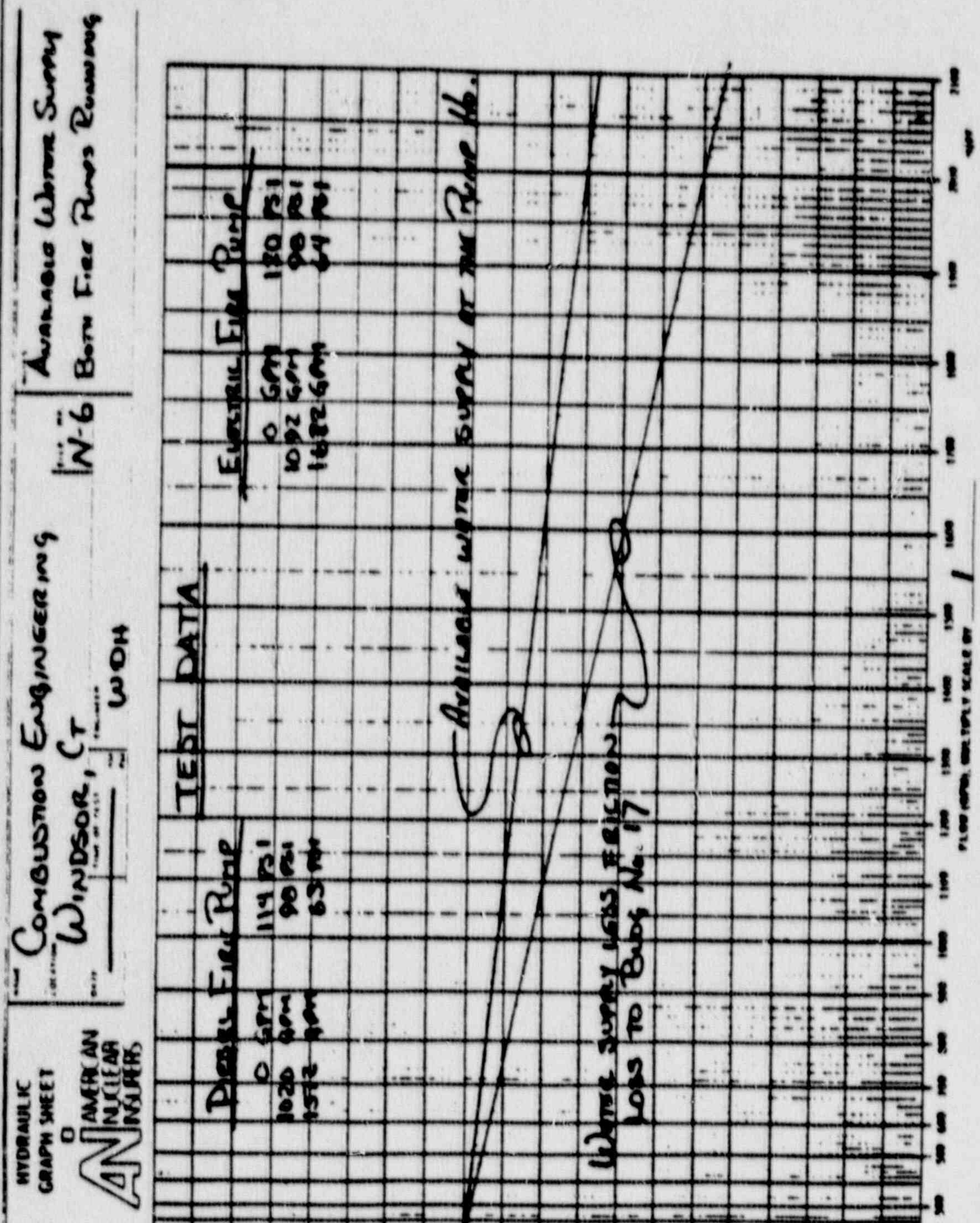

Figure 14-19

WATER SUPPLY TEST DATA FOR BUILDING NO. 17

Chapter 15 INTENDED POST-REDEPLOYMENT PROCESS DESCRIPTION AND SAFETY ANALYSES

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

a. the use of safe subcritical units either individually or in arrays, i.e., safe geometry in individual units or arrays,
b. a limit on the maximum enrichment of 5 w/o U-235 in all manufacturing processes, and
c. the use of favorable geometry in combination with engineered safety features to maintain process variables within a safe regime.

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

15.1 Process Outline and Moderation Control

Figure 15-1 outlines the major process steps between receipt of fuel pellets and shipping of completed fuel rod bundles. Autoclave testing is done only on selected rods; thus, most fuel rods bypass this process step. The manufacture of fuel assembly components other than elements containing SNM is not addressed here.

Moderation control is employed in the plant design and impacts the SNM processing in the following manner. Fire fighting techniques do not permit the use of high pressure hoses in any area where SNM is processed. All areas of Building 17 employ an overhead sprinkler system for fire fighting. As noted in Section 14.7, the volume fraction of water in the air produced by the sprinklers is 0.000075 and for analyses where the mist is pertinent, it is conservatively represented as 0.001 g/cc.

All criticality analyses include internal and external moderation where it is a credible condition. However, specific process steps, such as the fuel rod storage area, are enclosed to prevent water from entering the storage containers. Flooding of the SNM processing areas is not considered a credible event because of the building design and drain systems. Care has been taken to route water lines away from areas where water line rupture could spray on SNM processing operations. This is viewed as more of a radiological control than a criticality control problem.

15.2 Pellet Receiving

Description - Fuel pellets are received on-site in UNC-2901 shipping containers. Up to two shipping containers are attached to an individual pallet. Loaded shipping containers attached to pallets may be stored within the security fence, as discussed in Section 15.10.

Shipping containers on pallets enter Building 17 either via the conveyor along the western end of the north side of the building or by a forklift truck. Only one shipping container may be opened at a given time within the defined spacing area. Upon opening, the stack of pellet boxes is removed from the shipping container and placed upon a receiving table. The constraints on the stack of boxes are released and the boxes are unstacked and placed on a cart in a slab array which meets the safe slab requirements of Chapter 4.0. Upon removal of the contents of one shipping container from the spacing area, the next shipping container may be opened.

Safety Features - The shipping containers are demonstrated safe containers based on the analyses in the Certificate of Compliance. These analyses assume flooding of the container internals and full reflection. Inside of Building 17, the shipping container may be exposed to a water mist environment in the event that the sprinkler system should be activated. This is conservatively represented as a 0.001 g/cc water mist (see Section 14.7). Individual pellet boxes are covered and the likelihood of filling the boxes with water as a result of the sprinkler system is very low. The probability of flooding of the unloading area is negligible because of the building design, as noted in Section 15.1.

Nuclear Safety - As noted above, the pellets within the shipping container have been demonstrated to be safe in the analyses contained in the Certificate of Compliance. When the container is opened and a package of boxes is placed on the receiving table, it consists of 16 boxes forming a rectangular array 20.5 inches long by 10 inches wide by 8 inches high. Thus, this array exceeds the safe slab thickness of Table 4-2 by a factor of two. However, during the short time interval required to remove the constraints and place the boxes in a safe slab geometry, the probability for a criticality incident is negligibly small for the following reasons.

1) The probability for the sprinkler system to be activated at the instant that the stack of boxes is transferred to the receiving table is finite but small.

2) The fuel pellets are in boxes with covers; thus, the boxes are protected against water ingress from the sprinklers. The probability of flooding all 16 boxes is much less than the probability of flooding a single box.

3) The volume of the stack of stainless boxes (20.5"l x 10"w x 8"h) is 26.87 liters; this exceeds the reflected minimum critical volume of Figure 14-8 by 21% but is less than two-thirds of the minimum critical-unreflected volume at optimum moderation.

4) The UO2 pellets are randomly loaded into the boxes with a minimum of 7.94 Kg per box. Thus, the achievable water to oxide volume ratio is 1.32 which is less than 50% of the optimum value for a slab geometry.

The spacing area for this operation is set at a minimum of 25.0 square feet using a maximum box loading of 9.07 kg UO2 times 16 boxes divided by 5.8 kg $UO2/ft^2$ (see Table 4-3).

The nuclear criticality safety of the conveyor system for bringing UNC-2901 loaded pallets into Building 17 is based on the analysis for storage of these containers presented in Section 15.10. This analysis shows that an infinite planar array of shipping containers, when reflected top and bottom, had a multiplication factor below 0.70 over the full range of interstitial water density. Consequently, the conveyor system is safe when viewed as a portion of this infinite array. Figure 15-2 shows the location of the conveyor system and Figure 15-3 details the conveyor system geometry.

Transfer of shipping container pallets by forklift truck from storage areas outside the building to an unloading area closer to the pellet stacking tables is done via an access route through the machine and rod processing shop areas.

15.3 Pellet Storage and Transfer

15.3.1 Pellet Storage

Description - Covered steel shelves are provided for pellet storage. The storage area is three shelves high. Each shelf has a depth of 30" and is limited to a slab thickness of 4.0". The slab thickness of 4.0" is assured by limiting the number of fuel pellet boxes stacked at any position.

Safety Features - The entire storage array is covered by a sheet metal top which prevents significant moderation of the array from the discharge of the overhead sprinkler system. Fire fighting by high pressure fire hoses is not permitted in the pellet shop.

Nuclear Safety - The following assumptions are incorporated in the calculational model of the Pellet Storage Area:

1. All steel structural materials of the shelving are neglected.

2. An external mist of .001 g/cc is assumed.

3. Each of the three storage shelves is 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). The stainless steel boxes are explicitly represented.

4. A 0.25" film of water is 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) are also included.

6. The pellet storage shelves are modelled as a system of infinite length parallel to the concrete wall.

7. A twelve-inch thick water reflector is placed immediately in front of the shelves.

Figure 15-4 illustrates the pellet storage shelves.

The NITAWL and XSDRNPM codes are used to obtain 16-group cross sections from the 123-group super-XSDRN library for input to KENO-IV, the 3-dimensional code which was used to determine the reactivity of the pellet storage area under the conditions noted above. A keff of 0.8814 ± .0046 is obtained. Since this analysis employed a 12 inch thick water reflector immediately adjacent to the front of the storage shelves, a one foot exclusion area at the front of the shelves is acceptable.

15.3.2 Pellet Transfer

Description - UO2 may be transferred in approved containers by hand or on carts. The carts accommodate one mass limited container or a slab limited array of containers. This mode of transfer is also applicable to clean and dirty scrap as well as waste materials. Clean and dirty scrap UO2 shall be contained in approved containers. Waste materials such as contaminated rags, gloves, filters, etc. shall be packaged and transported according to applicable internal waste handling procedures.

Up to two fuel rod storage boxes may be transported on the fuel rod storage box transport cart.

Safety Features - The various hand operated transport devices for moving SNM about the facility are specially designed to meet criticality limits and controls. Slab limited carts for pellets and scrap are designed to position the slab at the fixed height above the floor, thus adjacent slab limited carts of this type do not require spacing since they form a single coplanar slab.

The slab limited cart for fuel rod storage box transfer is designed to provide the proper spacing from adjacent equipment by positioning of the fuel rod storage boxes in specific areas of the cart and providing barriers to limit the closest distance of approach.

Mass limited transport carts have a minimum size consistent with surface density spacing requirements and employ a mechanical positioning device to assure centering of the container within the spacing area.

Nuclear Safety - All approved transport devices for SNM material employ mass or geometric safe limits. Because most equipment spacing areas do not extend beyond the physical boundary of the equipment or mechanical barriers around the equipment, spacing between transfer carts and equipment is of no concern. In cases where the spacing area extends beyond equipment boundaries, such as 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 limited container or slab limited array of containers, and then only to permit an operator to transfer that material to an available storage position. SNM in transit on transfer carts shall not cross a spacing boundary unless it is entering that spacing area for processing or storage.

15.4 Fuel Rod Assembly and Inspection

Description - The pellet stacking/loading tables have a perforated flat surface. Troughs, approximately 3/4" x 3/4" x 12 ft long, are placed in a single layer on the surface, and UO_2 pellets are stacked in the troughs to a specified length. While on the stacking table the pellets are inspected and sorted to insure pellet quality requirements are met. Defective pellets are placed in shallow (less than 3" high) pans also located on the perforated surface.

Empty fuel tubes are brought to the station in fuel rod transfer carts which hold 250 tubes in the form of five parallel annular rings, fifty tubes to a ring. The innermost ring has a diameter of approximately 15-3/4" and the outer ring has a diameter of approximately 24-1/2". These fuel rod transfer carts are used to move and store fuel rods as they move from the stacking/loading tables to subsequent operations including end cap welding, deflashing, and fluoroscopy. Deflashing is performed on one rod at a time. During deflashing the rod normally stays in the cart, except that one end is drawn out of the cart far enough to be inserted into the deflasher.

Rods are removed from the carts for leak testing. Leak testing is performed on up to 20 rods at a time. Leak testing may be performed prior to or subsequent to enrichment verification and fluoroscopy. If leak testing is performed prior to enrichment verification and fluoroscopy the rods may or may not be returned to the rod transfer cart. If leak testing is performed subsequent to enrichment verification and fluoroscopy rods are not normally returned to the transfer cart, but are instead loaded directly into fuel rod boxes which have a maximum length of 172 inches, an inside width of eight inches and a height of 5-3/8 inches. The rods are typically packed in a close fitting hexagonal pitch array within the fuel rod box. The fuel rod boxes are then normally placed in the fuel rod storage area or moved to subsequent processing operations.

Rods are fed to and from the enrichment verification and fluoroscopy operation using an inclined plane.

Safety Features - The pellet loading table surface is perforated; the underside of this surface operates at a negative pressure to collect dust and small chips which may pass through the perforations and thus minimize UO_2 dust in the controlled climate of the rod loading room. All SNM operations, including pellet stacking, pellet chip and broken pellet accumulators, and spare pellets, are done within safe slab limits.

Upon completion of the inspection operations, the completed rods are packed in a fuel rod storage box in a close fitting hexagonal array; filled storage boxes are placed in the fuel rod storage area. This storage area is completely enclosed to prevent water from any potential source from contact with the fuel rods.

Nuclear Safety - As noted in the previous paragraph, all SNM handling in the rod loading operation is done within the safe slab limits defined in Chapter 4.0.

The fuel rod transport carts are analyzed explicitly to assess their criticality safety. The following assumptions are incorporated into the calculational model of the Fuel Rod Transport Carts:

1) Only the 1/4 inch thick, 8" O.D. steel structural support tube for the annular fuel rod storage region is accounted for in the model. See Figure 15-5. All other construction material is neglected.

2) The carts are assumed to be an infinite close packed array in the horizontal plane.

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

4) Each fuel rod is contained in a 133.5 inches long 1/2 inch Schedule 40 PVC tube. There are 250 tubes arranged in 5 concentric rings with an average radial pitch of 1.10 inches and azimuthal pitch of 7.2 degrees. The fuel tube region of the cart is thus a cylindrical annulus 7.4 inches from the center of the cart and extending to a radius of 12.7 inches. Above this annulus and at each edge of the cart is a weld sample box (4 3/8" x 4 3/8") 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 viewing areas encloses the top, sides, and back of the cart. In the calculational model it is assumed that all 250 positions in the annular storage area and all 50 positions in the weld sample boxes are occupied by the largest diameter rods (0.3765" O.D. UO2 pellets at 10.061 gm/cc stack density with a Zr-4 cladding thickness of .028 inch) at the maximum UO2 enrichment of 5.0 wt % U-235. It is also assumed that the fuel rods and the PVC tubes extended the full length of the cart (165 inches). The moderation effect of the PVC is included in the analysis, however, the absorption effects are been neglected. A 0.25" film of water is assumed on the exterior sides of the cover. The concrete floor and ceiling are also modelled.

The NITAWL and XSDRNPM codes are used to obtain 16-group cross sections from the 123-group super - XSDRN library for input to KENO-IV. The resulting multiplication factor for the Fuel Rod Transfer Cart array, based on the conditions described above, is 0.873 $\pm$0.0058.

Guard rails prevent the carts from coming any closer than three feet center-to-center. This provides a minimum separation between the outer edge of the fuel annulus and the edge of the cart of 5.3 inches.

The above analysis assumed the longitudinal axes of the transport carts were parallel in the semi-infinite array. This analysis is conservative since the analysis was done assuming infinitely long fuel rods surrounded by the PVC tubes whereas these regions do not extend the full length of the cart; the PVC tube is 133.5 inches in length, the maximum fuel column length is ~150 inches, and the length of the cart is 166 inches. If carts are abutting at right angles, the configuration is less reactive since the distance between adjacent moderated regions is increased from 10.6 to 21.3 inches. The 21.3 inches consists of the 5.3 inch distance between the side of the fuel annulus and the boundary of the cart plus the 16 inches between the ends of the PVC tubing and the cart. The latter 16 inches may consist of up to approximately 12 inches of dry (unmoderated) clad fuel pellet plus 4 or more inches of air at the end of the cart. The neutronic coupling between the PVC moderated regions of the two carts afforded by the 12 inches of unmoderated UO2 plus 9.3 inches of air is less than that afforded by the 10.6 inches of air when the axes of the carts are aligned.

15.5 Autoclave Corrosion Test

Description - Two autoclaves are employed for corrosion testing of finished fuel rods. The stainless steel autoclaves are 14 foot deep vertical tanks mounted with their tops approximately 30 inches above the floor. The tanks have an inside diameter of 14 inches, a wall thickness of 1.5 inches, and a minimum center to center distance of 66 inches. Each autoclave is limited to 32 fuel rods by administrative control.

Nuclear Safety - Each autoclave is limited to 32 fuel rods by administrative control. During operation, the interior of the autoclave experiences a broad range of moderator density conditions. Therefore, the criticality safety of a cluster of fuel rods in an autoclave can be conservatively assessed by assuming optimum moderation conditions and using the cylinder data of Figure 14-15 for 0.1 inch diameter pellets. A concentration of 2.5 gUO2/cc is equivalent to a volume ratio of water to UO2 of 3.02 based on a UO2 stack density of 10.06 g/cc. Thus each fuel rod and associated water occupies an area of 4.02 times the pellet area. For 0.4 inch pellets, the effective diameter for the 32 rod cells is 4.54 inches. Should the autoclave be double batched, the equivalent cylinder diameter is 6.41 inches. Both diameters represent safe cylinder diameters using the data of Table 4-1.

Interaction between the autoclaves is minimal because of the 66 inch center-to-center spacing and the nearly full reflection afforded by the radial water reflector plus the 1.5 inch stainless steel autoclave wall (see Figure 14-12).

15.6 Fuel Rod Storage Area

Description - The multi-level storage area shown in Figure 15-6 for boxes of fuel rods consists of four vertical 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 approximately 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 an average spacing of 4 inches. The rod boxes rest on roller conveyers to facilitate movement in and out of the storage array.

Safety Features - The entire storage array is covered by sheet metal to assure the exclusion of sprinkler water. The fire resistant top cover 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.

Nuclear Safety - The following assumptions are incorporated in the calculational model of the Fuel Rod Storage Area depicted in Figure 15-6.

1) Each of the rod boxes is assumed to contain 0.382" OD fuel rods at an enrichment of 5.0 wt % U235 with UO2 stack density of 10.061 g/cc. The fuel rods are assumed to be tightly packed in a hexagonal array. The 8" wide tray is filled to a height of 6.25", which is greater than the box height of 5.375 inches and the slab limit (6 inches) specified in Table 4-2, for a total of 371 fuel rods. The fuel is assumed to be dry. The fuel and clad are homogenized over the volume of the tray.

2) A vertical spacing of 11.5" rather than 12.5" between rod boxes is assumed.

3) A lateral separation distance of 3.5 rather than the average of 4 inches between rod boxes is assumed. Interspersed moderation is not considered credible since moderation control is assured by the cover, walls, and doors of the storage area.

4) All steel construction material is neglected.

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

6) The one eighth inch thick rubber pad at the bottom of the tray, as shown in Figure 15-6, is modeled as water in the bottom of the box.

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

15.7 Fuel Rod Pre-Stacking Station

Description - The fuel rod pre-stacking station is employed to load and log serial numbers for fuel rods placed in a rod box in a configuration corresponding to that for the completed fuel assembly. Each layer of rods in the box corresponds to a given row in the fuel assembly. Fuel or other rods may be extracted from one of three rod boxes on the positioning table, which is a device used to vertically align one of the three boxes on the positioning table with the box being loaded. CEA guide tube locations are included in the rod array within the box being loaded by placing clusters of four empty clad tubes. The rods in the loaded box are layered in a close packed hexagonal array. Upon completion of the loading of the pre-stacked box, it is transferred to one of the two special locations in the Fuel Rod Storage Area which are aligned with ports opening through the wall to the locations where the rods are inserted within the fuel assembly cage structure.

Nuclear Safety - The following assumptions are incorporated in the calculational model of the Fuel Rod Pre-Stacking Station.

1. The three boxes on the positioning table are stacked vertically with a distance of 8.5" between the bottom of one box and the bottom of the next. Each box is assumed to contain the largest diameter fuel rods (0.44"OD)at an enrichment of 5.0 wt % U235. The fuel rods are assumed to be tightly packed in a hexagonal array. The boxes are conservatively assumed to be 9" wide, filled to a height of 6.1" and containing 312 fuel rods each. Each box is conservatively assumed to be flooded with water. The fuel, clad, and water are homogenized over the volume of the box.

2. All structural materials employed to support the fuel rods are neglected.

3. An external mist of 0.001 g/cc is assumed.

4. A concrete ceiling (4") and floor (16") are included in the calculation.

The NITAWL and XSDRNPM codes are used to obtain 16-group cross sections from the 123-group DLC-16 library for input to KENO-IV, the code which is used to determine reactivity of the Fuel Rod Pre-Stacking Station under the conditions noted above. A k_{eff} = 0.8475 ± .0054, is obtained for an infinite array of stations 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 CEA positions consisting of 4 empty clad tubes per position. Since the array has a much lower water to fuel ratio than a fuel assembly, even if the 20 fuel positions were filled with water within the array, the reactivity would be less than that of a fuel assembly. The fuel rod box with the pre-stacked array when placed in the rod storage area will be less reactive than the array analyzed in Section 15.6 in that the pre-stacked fuel rod box has fewer fuel rods per box, 239 (16 x 16) or 176 (14 x 14) fuel rods versus 371 fuel rods assumed for a loaded box in Section 15.6. 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 box which is pulled partially into the fuel assembly room is again less reactive than a fuel assembly. Only two fuel rod storage boxes can be partially pulled into the room in fixed positions which are separated by over 4 ft.

15.8 Fuel Assembly Fabrication

Description - Fuel and other rods, if any, are transferred from the prestacked rod box to the fuel assembly loading table one row at a time. Each row is guided into the fuel assembly cage structure by a fixture which provides a lubricating water spray at each grid location. The structure of this equipment is designed to prevent retention of water in the fuel region of the assembly. Table 15-1 summarizes pertinent parameters for the two typical C-E fuel assemblies. Other fuel assemblies manufactured to date are less reactive than those of Table 15-1.

Safety Features - The fuel rod box and fuel rod assembly are at a working level well above the floor. Thus, flooding of the fuel is not likely since the primary sources of water are the water spray on the grid cage structure and the overhead fire sprinkler system. Water accumulation in the room is prevented by large drains at floor level to the machine shop area and the floor sump noted in Section 15.6.

Nuclear Safety - Nuclear Safety of the fuel assembly fabrication operation is based on the following. The pre-stacked rod box projects through the wall from the rod storage area by only a short distance so as to permit grasping of the fuel rods in a single row. The fuel rods are pulled onto the loading table where they form a slab one rod thick. The partially or fully loaded fuel assembly cage is very highly under moderated and separated from the pre-stacked tray by over ten feet. The nearest SNM to the fuel assembly structure portion of this operation is the adjacent loading operation which is over six feet away. Since the number of fuel rods in close proximity to the cage structure being loaded is limited to the number of rods in the completed assembly, this operation is no more reactive than a partially moderated and partially reflected fuel assembly; i.e., $k < 0.92$ (cf. Table 15-1).

15.9 In - Plant Storage of Fuel Assemblies

Description - Fuel assemblies are stored in a vertical position using racks of adequate strength to preclude loss of the design spacing. There are a maximum of 180 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 fuel rods are loaded into the fuel assembly cage structure, a vertical wash station where each assembly receives 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 assemblies are loaded into shipping containers prior to outdoor storage. The fuel assembly wash station is an oversize shower stall with vertical strings of mist nozzles along the walls. 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 205 fuel assemblies, up to 180 storage positions, plus 25 additional locations. All assemblies outside of shipping containers are stored vertically within the design spacing criteria of the Assembly Storage Room. Figure 15-7 shows the Fuel Assembly Storage Room layout.

Safety Features - The fuel assembly storage room is sprinkler equipped for fighting fires. The base of the wall contains several ports for draining the room should water collect on the floor. Thus, flooding of the room is not possible. Individual fuel assemblies in the storage rack may have a plastic tubular sheeting pulled over the assembly for cleanliness. This tubing is open at the bottom to preclude collection of water within the tube.

Nuclear Safety - The following analysis of the fuel assembly storage rack was carried out to evaluate the multiplication factor for a 20 x 34 array of fuel assemblies, which is a very conservative representation of up to 180 fuel assemblies in the storage rack and the additional 25 fuel assemblies at various other locations in the room. See Figure 15-7 for the Fuel Assembly Storage Room layout.

The assumptions employed in this 20 x 34 array analysis are as follows.

1. The center-to-center spacing of fuel assemblies in each of the 20 rows is 9.75 inches versus 10.0 actual. The distance between pairs of rows not separated by an aisle is 35 inches center-to-center, and for rows separated by aisles, 37 inches center-to-center.

2. All steel construction material of the storage racks is neglected.

3. The water mist from the fire sprinkler system has been calculated in Section 14.7 to be 0.000075 grams per cubic centimeter. For conservatism, a water-mist density of 0.001 grams per cubic centimeter is assumed to be in and around the fuel assemblies in the storage array. This is a factor of about 13 times higher than the calculated mist density for the sprinkler system. A uniform water film thickness of 0.025 cm. is assumed on the fuel assembly surfaces. This exceeds the film thickness calculated in Section 14.7 by a factor of 2.6 after including the 15 percent uncertainty in the film thickness calculation.

4. The KENO model employed 8 inch thick concrete walls, a 16 inch thick concrete floor, and a 4 inch thick concrete ceiling surrounding the 20 x 34 array.

Sixteen broad group cross sections are collapsed from the 123 group DLC-16 library using the NITAWL and XSDRNPM codes. The KENO IV analysis of the storage array yields a multiplication factor of 0.842 $\pm$ 0.004.

15.10 Shipping Container Storage

15.10.1 927A1/C1 Shipping Container Storage

Description - Fuel assembly bundle shipping containers (Model 927A1 and 927C1), each containing up to two fuel assemblies, may be stored in arrays up to three high without violating criticality evaluations discussed below. The storage array width and length are limited only by the space allocated for storage. The steel shipping container, approximately 3 feet in diameter and up to 217" long, houses up to two fuel assemblies of the types described in this license. Figure 15-8 shows the dimensional details of the shipping containers. The two assemblies in each container are separated by six inches. An eight foot high chainlink fence encloses the storage area.

Safety Features - The Model 927A1 and 927C1 fuel assembly shipping containers are approved for shipping 14 x 14 and 16 x 16 fuel assemblies containing UO2 enriched up to 5 w/o U-235.

Nuclear Safety - Fuel assembly shipping containers, each with two fuel assemblies having UO2 of up to 5w/o U-235, may be stored three high in an infinite planar array. This statement is based on a KENO analysis employing the following assumptions.

1. An infinite planar array of shipping containers stacked three high. In the longitudinal direction of the shipping containers, the array composition is conservatively approximated as infinitely long fuel assemblies and containers.

2. The top and bottom of the semi-infinite planar array are each reflected by twelve inches of water.

3. All shipping containers are assumed to contain two 16 x 16 fuel assemblies, no burnable poison shims, and UO2 enriched to 5w/o U-235. In addition, all containers are assumed to be fully flooded with water internal to the container.

The NITAWL and XSDRNPM codes are employed to derive 16 broad neutron group cross sections from the 123 group DLC-library. The resulting libraries are used in KENO-IV to calculate effective multiplication factors for the array described above. A multiplication factor of 0.9202 ± 0.0075 is obtained.

The use of fully loaded and flooded shipping containers is highly conservative for array storage. However, it does establish a basis for considering the loaded fuel assembly shipping container as a safe subcritical unit, even when flooded, within close packed storage arrays or individually within the fuel fabrication facility.

15.10.2 UNC-2901 Shipping Container Storage

Description - The UNC-2901 shipping container consists of an outer shell having the dimensions of a 55 gallon drum (22.1 inch diameter by 35.6 inches high). Internally, the shell contains a sealed container 10.75 inches square by 30 inches long. This inner container is supported by structural members; insulation is also present between the inner container and the shell. The inner container may be loaded with a cage structure holding two cylindrical stainless steel containers of approximately 3.5 gallon capacity each. Alternatively, another cage structure may be employed to hold 16 covered pellet pans.

Two shipping containers are mounted horizontally on a pallet. Each pallet is treated as a shipping container unit. Loaded pallets may be stored in arrays up to three pallets high within the security fence, including Building 21, without violating criticality evaluations discussed below. The number of 2901 pallets within Building 17 is limited to four in the pellet loading area or three at the head of the 2901 conveyor system subject to the requirements that they be spaced at least one foot from process equipment.

Safety Features - The UNC-2901 shipping container units are approved for shipping UO2 pellets, hard clean scrap, and powder for U-235 enrichments, up to 5 w/o.

Nuclear Safety - UNC-2901 shipping containers may be stored within the security fence, as noted above, in arrays up to three high. These shipping containers may contain either 16 pans of uranium pellets or two uranium powder pails. The UNC-2901 shipping container is a 55 gallon steel drum with a diameter of 22.1 inches and a height of 35.6 inches.

The following assumptions were employed for the criticality evaluation of the UNC-2901 shipping container:

1) The array of contiguous containers is of infinite extent in the x-y plane, the major axes of the containers are in this plane and the height of the array is equivalent to three times the diameter of a shipping container (containers stacked on a square pitch).

2) In the finite (vertical) direction, the array is bounded by a twelve-inch concrete slab on the bottom and a twelve-inch thick, full density water reflector on the top.

3) The assumed water content in uranium pellets is 1.0 weight percent.

4) The assumed water content in uranium powder is 5.0 weight percent.

5) Except for the assumed water content in the uranium fuel, the inside of the container is completely dry.

6) The shipping container may be loaded with either 16 pans of pellets (20 pounds of uranium oxide per tray) or two pails of uranium powder (35 kilograms of uranium oxide per pail).

7) Water density in the interstitial area of the contiguous array of cylinders was treated as a variable in the KENO analysis.

Several KENO models are defined for the UNC-2901 container; one for the shipping container with 16 trays of pellets, one for the shipping container with two powder pails and wood structural members and one with powder pails and no wood structural members. The specular reflection option of KENO is employed to model the infinite horizontal array of shipping containers. A vacuum boundary condition is applied at the top and bottom of the axial reflector regions. The Hansen and Roach cross section library is employed to model the uranium powder pails. The XSDRN/NITAWL methodology is employed to generate a 16 group cross section library for the uranium pellets in the UNC-2901 container.

KENO calculations are carried out for UNC-2901 shipping container arrays containing either pellets or powder over a range of water density between the shipping containers. Since wooden structural members are present within the shipping containers, their effect on multiplication factor is also examined.

Figure 15-9 shows the variation of KENO eigenvalue versus water density for either pellets or powder without the wooden structural members and the case of pellets with the wooden structural members. These curves exhibit a broad maximum in the range of 0.1 to 0.2 g/cc water density. The maximum eigenvalue for the case of UNC-2901 shipping containers loaded with pellets and wooden structural members represented in the KENO model is less than 0.60.

15.10.3 Mixed Storage of 927A1/C1 and UNC-2901 Shipping Containers

Nuclear Safety - As discussed in Section 15.10.1, the effective multiplication factor for the models 927A1 and 927C1 fuel assembly shipping containers is less than 0.93. This analysis was done for a highly conservative array representation, i.e., a fully moderated system, of these containers. In storage, these containers would normally be dry on the inside and thus have a lower eigenvalue.

The effective multiplication factor for the infinite planar array of UNC-2901 shipping containers is, for purposes of this discussion, below 0.70. Thus, mixed storage of 927A1/C1 and UNC-2901 shipping containers is bounded by the nuclear characteristics of the 927 A1/C1 shipping containers and a mixed array should have a multiplication factor less than 0.93.

15.11 Recycle Operations

15.11.1 Fuel Rod Salvage

Description - 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.

Safety Features - Ventilated hoods have a minimum air flow across open faces of 100 cfm. Zirc chips are salvaged and stored according to internal Industrial Safety policies.

Nuclear Safety - All fuel rods and salvaged pellets are handled according to the nuclear safety limits defined in Chapter 4.

15.11.2 Clean Hard Scrap UO2

Description - Clean hard scrap is defined as broken pellets, pellet chips, and fines salvaged from various inspection steps up to loading pellets into clad tubes and fuel rod salvage operations. Each of these inspection operations and the salvage operations are performed under controlled conditions with a minimum likelihood of contamination of the fuel so as to degrade quality. This clean scrap is collected in safe geometry containers and placed in approved storage locations until shipment to the Hematite, MO. plant in approved shipping containers for reprocessing or other disposition.

Safety Features - As noted above, clean hard scrap is collected in safe geometry containers. All transfers between containers are carried out in ventilated hoods to minimize the potential for dusting.

Nuclear Safety - Safe geometry containers are employed for collecting clean hard scrap at various inspection and salvage steps in the overall operation. When trays are employed as scrap receptacles, they are stored in safe slab geometry prior to transfer to a hood or approved scrap storage locations. Transfer of the clean hard scrap to shipping containers is done within a mass limited hood. If 35 kg mass limited stainless steel cans are being employed as the basic unit in the approved shipping container, approved procedures for filling, closing, labelling, and subsequent handling of these cans are employed. All filled cans are closed prior to removal from the hood, transferred in approved vehicles, and stored in approved storage areas.

15.11.3 Pretreatment of Low Level Liquid Wastes

Description - All liquid waste water from mop buckets and other liquid cleaning operations in the contaminated portion of the shop is pumped into the liquid waste settling tank, a nearly horizontal slab geometry tank. This tank is part of a closed system consisting of a high efficiency centrifuge and the slab storage tank. Liquid circulates from the storage tank through the centrifuge and back to the tank. This process is continued until the level of contamination is low enough, as determined by sampling, for discharge to the retention tanks in Building 6. The centrifuge bowl has a capacity of 19 liters.

Nuclear Safety - The closed centrifuge system consists of the centrifuge and the slab storage tank. The centrifuge bowl is a 19 liter volume which is a safe volume container at 5 w/o U-235. The slab storage tank is four inches thick and supported in a horizontal plane approximately two feet off the floor. The square container has a mesh barrier around it so as to prevent any significant moderator material from approaching either slab face. Consequently, the slab storage tank is a safe, partially reflected container. Solids from the centrifuge are placed in safe geometry containers, dried in an oven within the hood, and then stored for further disposition. Hard, dirty scrap is handled under the SIU mass limits defined in Table 4-2.

Surface area spacings are defined for various regions of the centrifuge hood station using the criteria of Chapter 4.

15.11.4 Concrete Block Storage Area

Description - A vertical concrete block storage area is provided for the storage of 5 gallon, or less, metal containers containing up to 35 kg UO2 hard clean scrap, SIU mass limited quantities of dirty scrap, or other SNM materials meeting appropriate mass limits.

There are five vertical storage shelves at each location (see Figure 15-10) along the concrete wall; each storage location is separated from adjacent locations by 10 inches of concrete. Steel shelves of at least 16 gauge thickness are built in the structure with a vertical spacing of at least 16 inches. Each storage location measures 16 inches wide and 14 inches deep, and is lined by 1/4 inch thick steel on the two sides and back. The face of each storage location is secured by a wire mesh door.

Safety Features - The wire mesh door permits easy observation of the contents of each storage compartment and the steel lining permits easy cleaning of the storage compartments.

Nuclear Safety - The following criticality safety analysis demonstrates the acceptability of a twelve-inch exclusion area at the front of the Concrete Block Storage Area. The following assumptions are incorporated in the calculational model of the Concrete Block Storage Area.

1. All steel structural materials are neglected.

2. An external mist of .001 g/cc is assumed.

3. Each storage position is assumed to be occupied by a 5-gallon steel container containing a homogeneous mixture of 35 kg UO2 at 5.0 wt % and 5 wt. % H2O. This mixture is assumed to be uniformly distributed within the container.

4. A 0.25" film of water is assumed on the exterior steel walls of the shelving, the top of the shelves, and the exterior of each bucket.

5. A twelve-inch thick water wall is placed at the front of the Concrete Block Storage area.

The KENO-IV code with sixteen group Hansen-Roach cross sections is used to determine the reactivity of the Concrete Block Storage Area under the conditions noted above. A Keff of 0.4698 ± 0.0070 is obtained for an infinitely long array of storage positions.

An additional calculation assumes each storage container is completely flooded internally 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 shelves and liners. This latter analysis eliminates the need for a moisture analysis of the contents of each container prior to placing the container in the concrete block storage area.

15.11.5 Filter Knockdown Hood

Description - The filter knockdown hood is a ventilated glove box hood employed to remove loose UO2 powder from absolute filters and prefilters.

Safety Features - This is a closed hood to preclude any moisture ingress from the sprinkler system.

Nuclear Safety - The filter knockdown hood is a 35kg mass limited hood. However, the SNM removed from filters is treated as dirty scrap. Consequently, it is handled as dirty scrap in preparation for shipment to the Hematite Facility for reprocessing.

15.11.6 Reduction Furnace

Description - A dewaxing furnace will be employed for burning off volatile materials from scrap UO2. Material will be charged in sintering trays in a in-line slab array of less than 4 inches.

Nuclear Safety - This operation meets the SIU limits of Chapter 4 since the material will meet the requirements of randomly loaded heterogeneous material if the material consists of pellets and chips.

15.12 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.

Table 15-1 Typical Design Parameters of C-E Fuel Assemblies

Fuel Assembly		
Fuel Rod Array per Assembly	14 x 14	16 x 16
Total No. Fuel Rod Positions per Assembly	176	236
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 Pellets		
Material	UO_2	UO_2
Diameter, in.	0.3765	0.325
Length, in.	0.450	0.390
Density, % Theoretical	94 to 96.5	94 to 96.5
Spacer Grid		
Material	Zr-4	Zr-4
No. per Assembly	≤ 8	≤ 11
K_{eff} (5 w/o U-235, isolated in room Temp. H_2O)	0.92	0.91

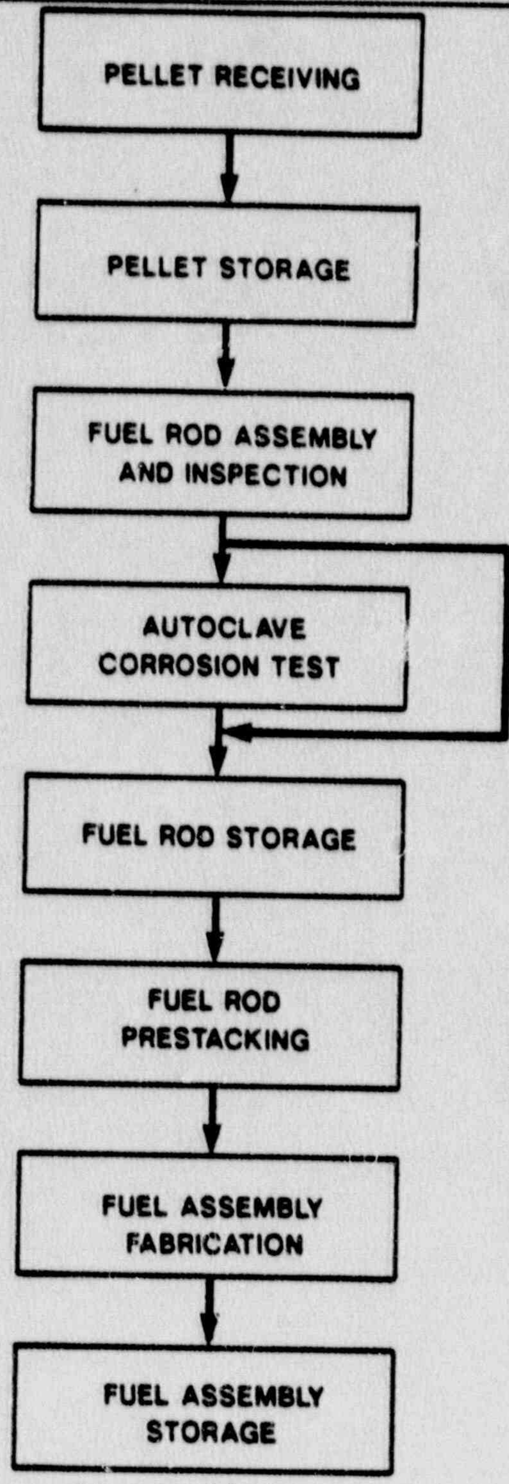

Figure 15-1

Process Outline

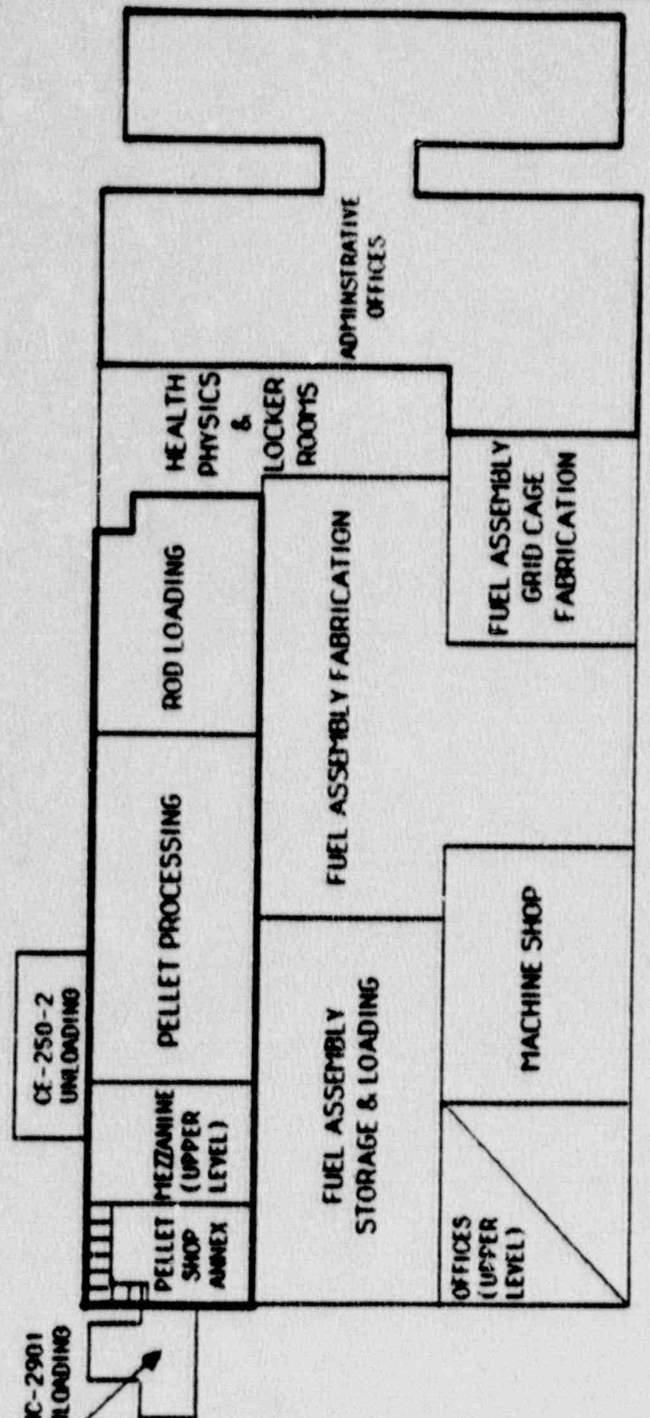

Figure 15-2
Building 17 Layout

PART II
SAFETY DEMONSTRATION

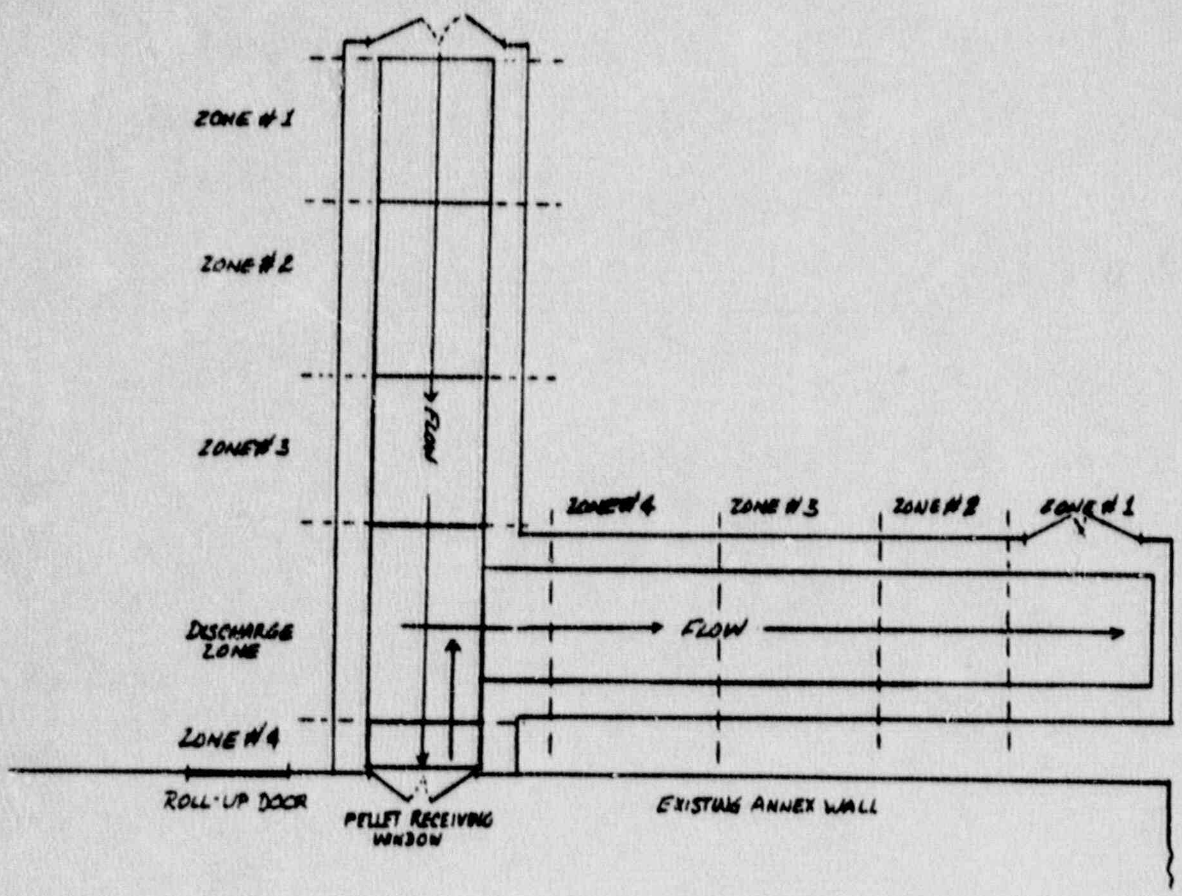

Figure 15-3
2901 Conveyor System

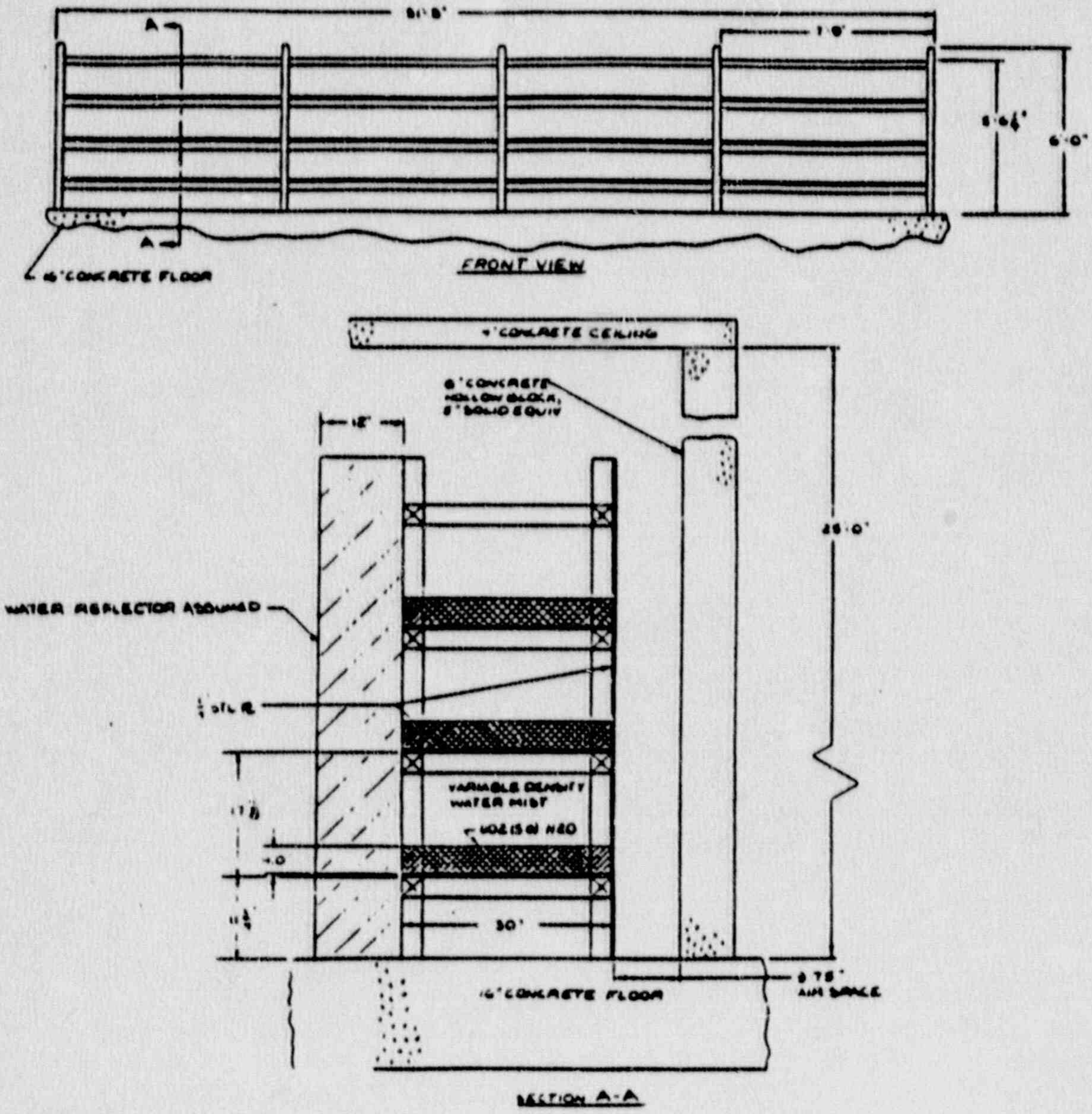

Figure 15-4
Pellet Storage Shelves

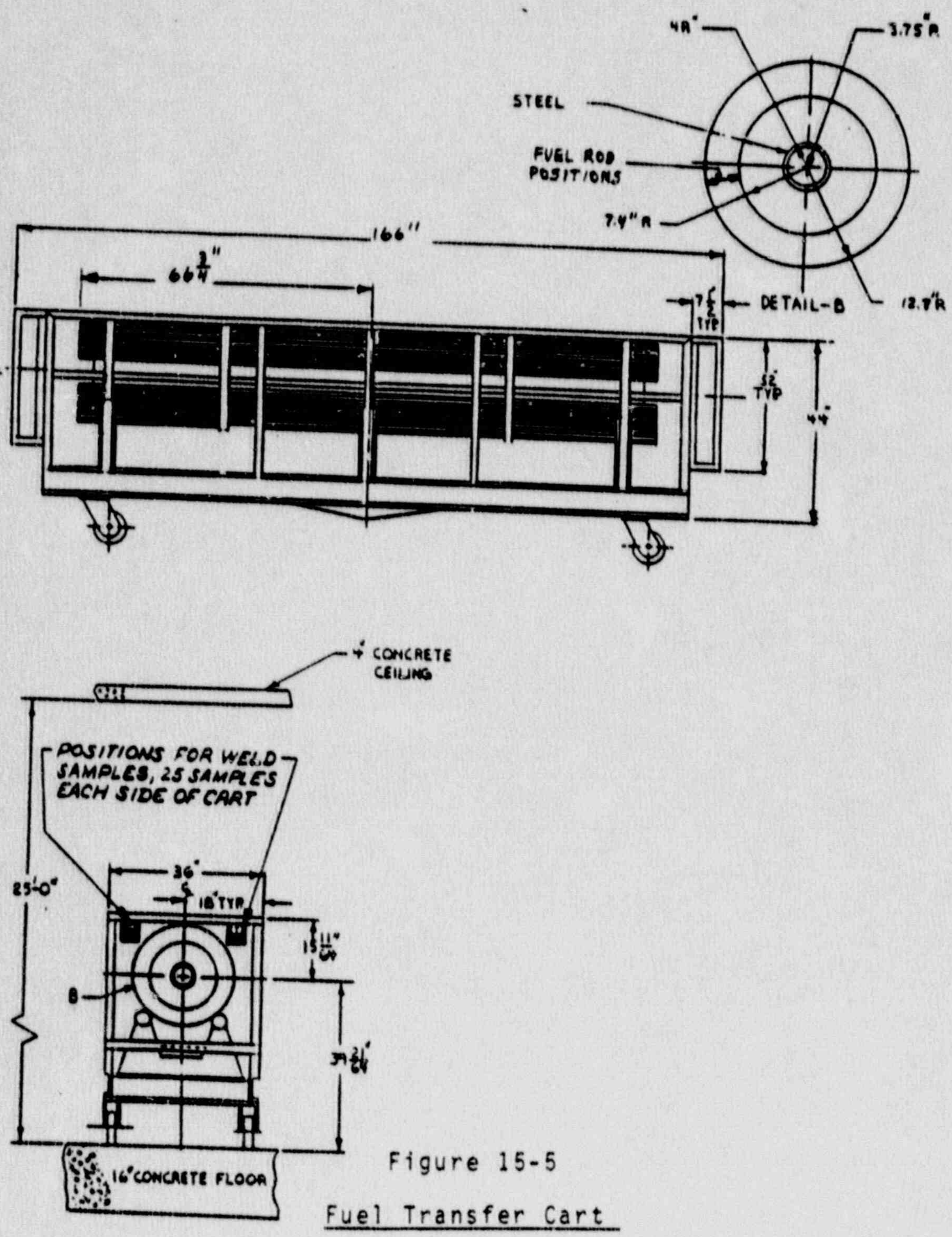

Figure 15-5

Fuel Transfer Cart

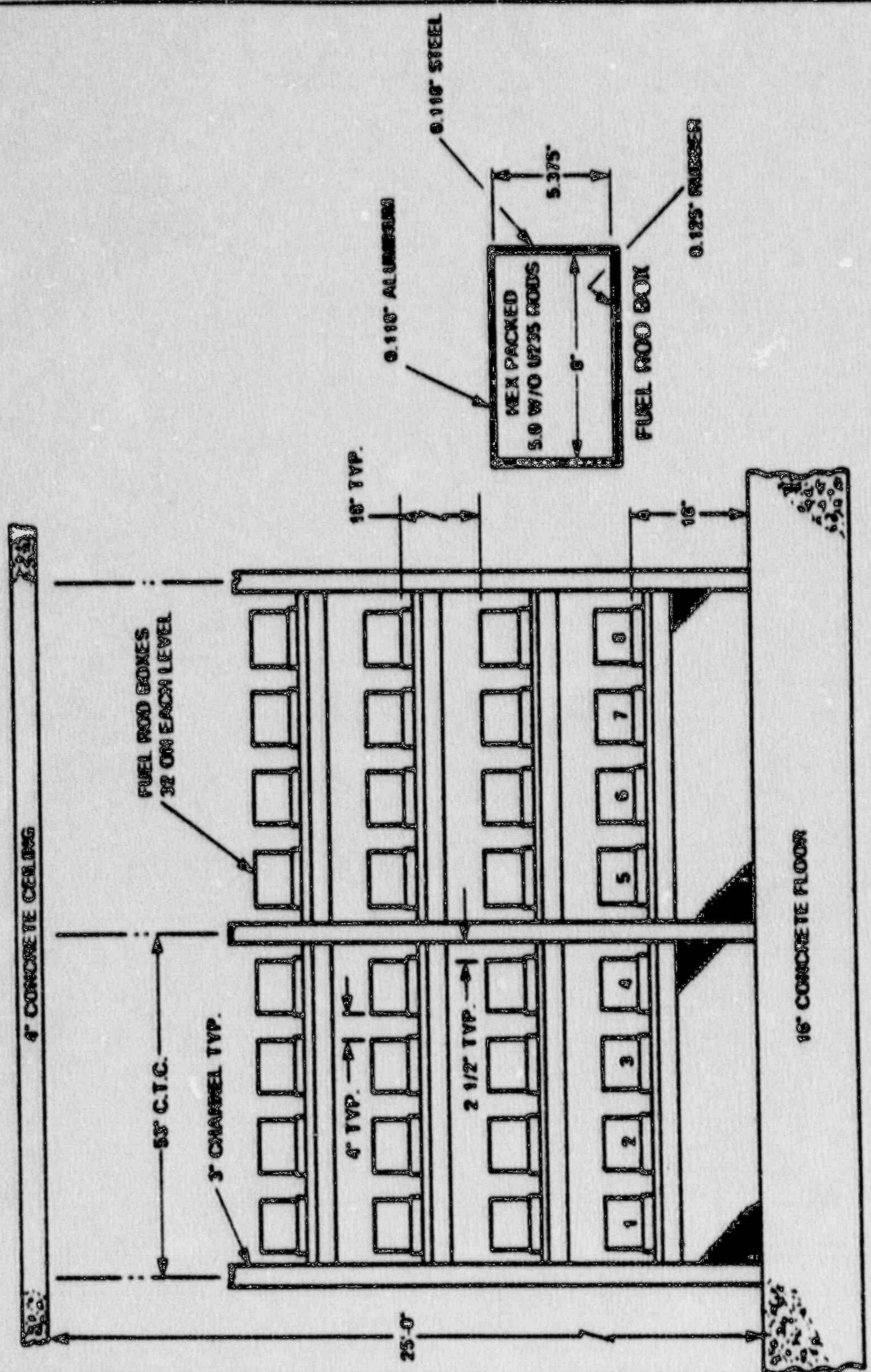

Figure 15-6

Rod Box Storage Area

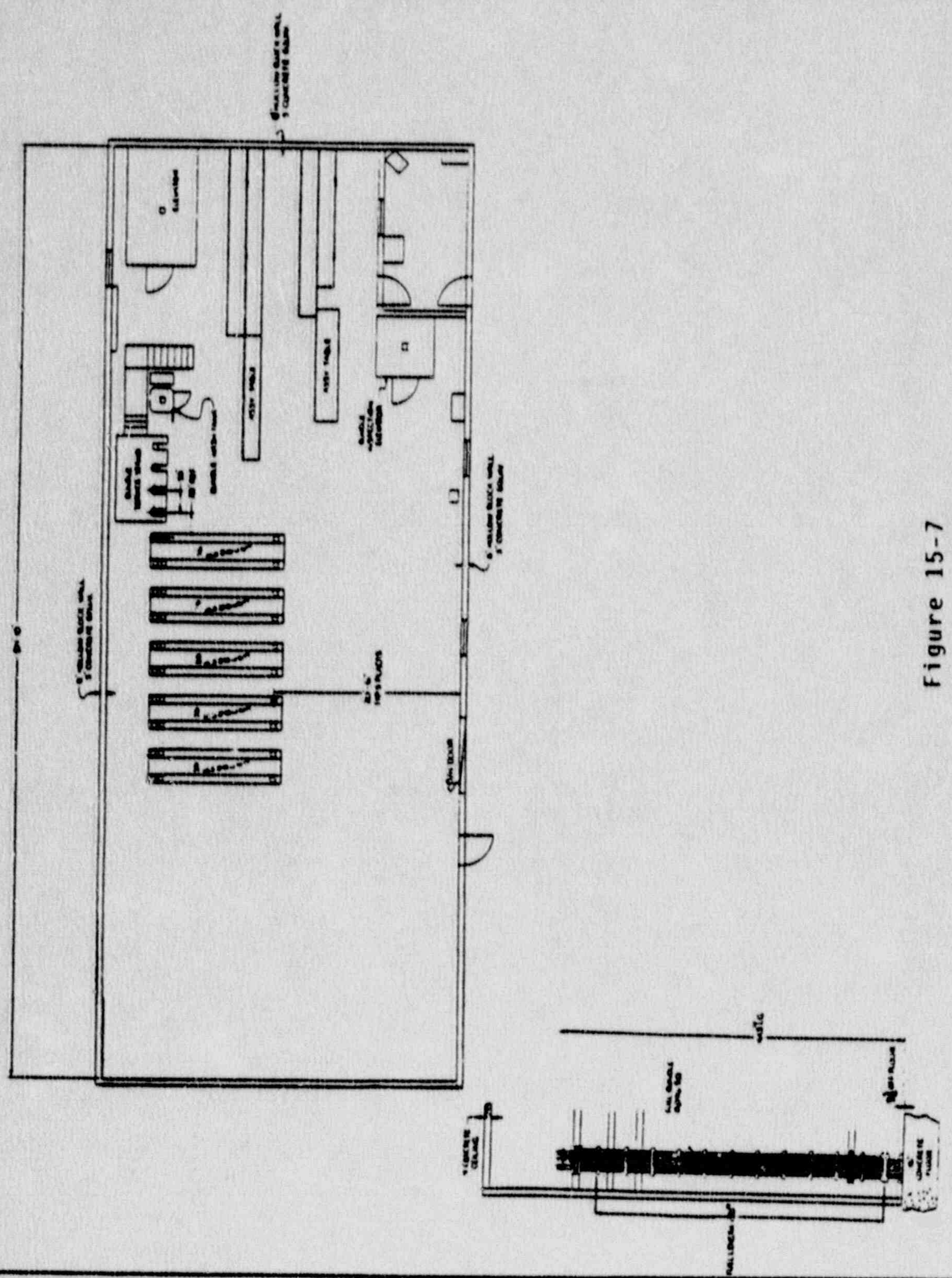

Figure 15-7

Criticality Model, Fuel Assembly Storage Room

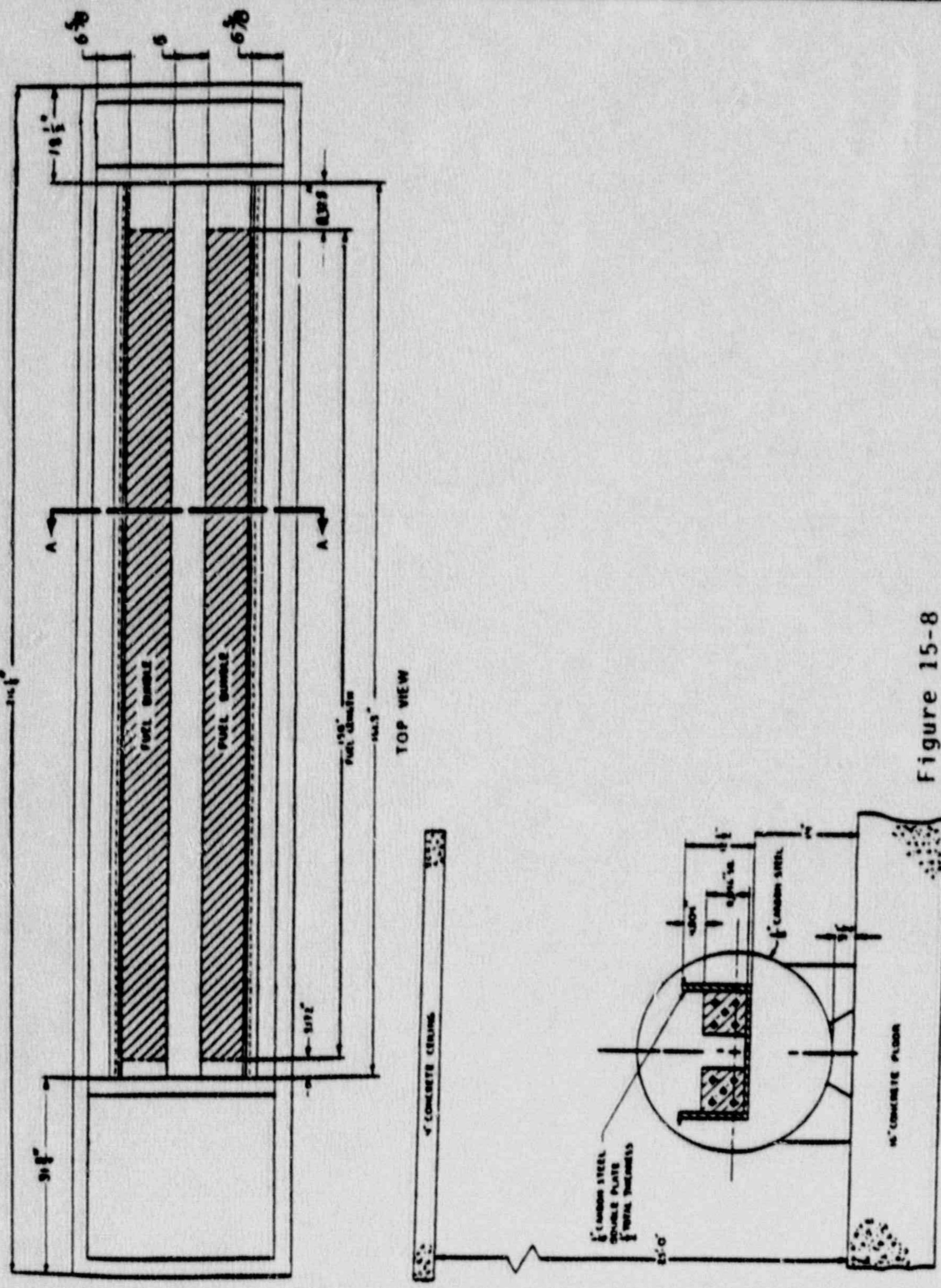

Figure 15-8

Fuel Assembly Shipping Container

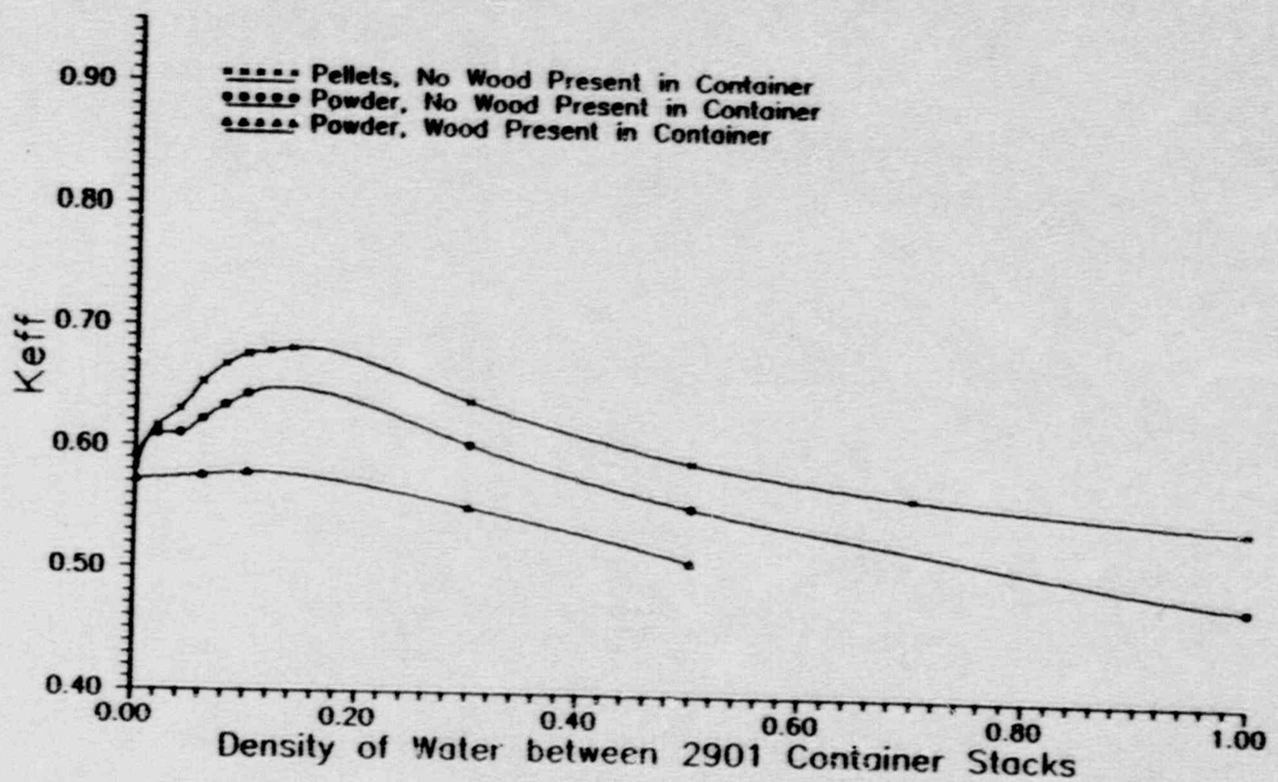

Figure 15-9
K_{eff} Of 2901 Container Array vs Water Density Between Container Stacks

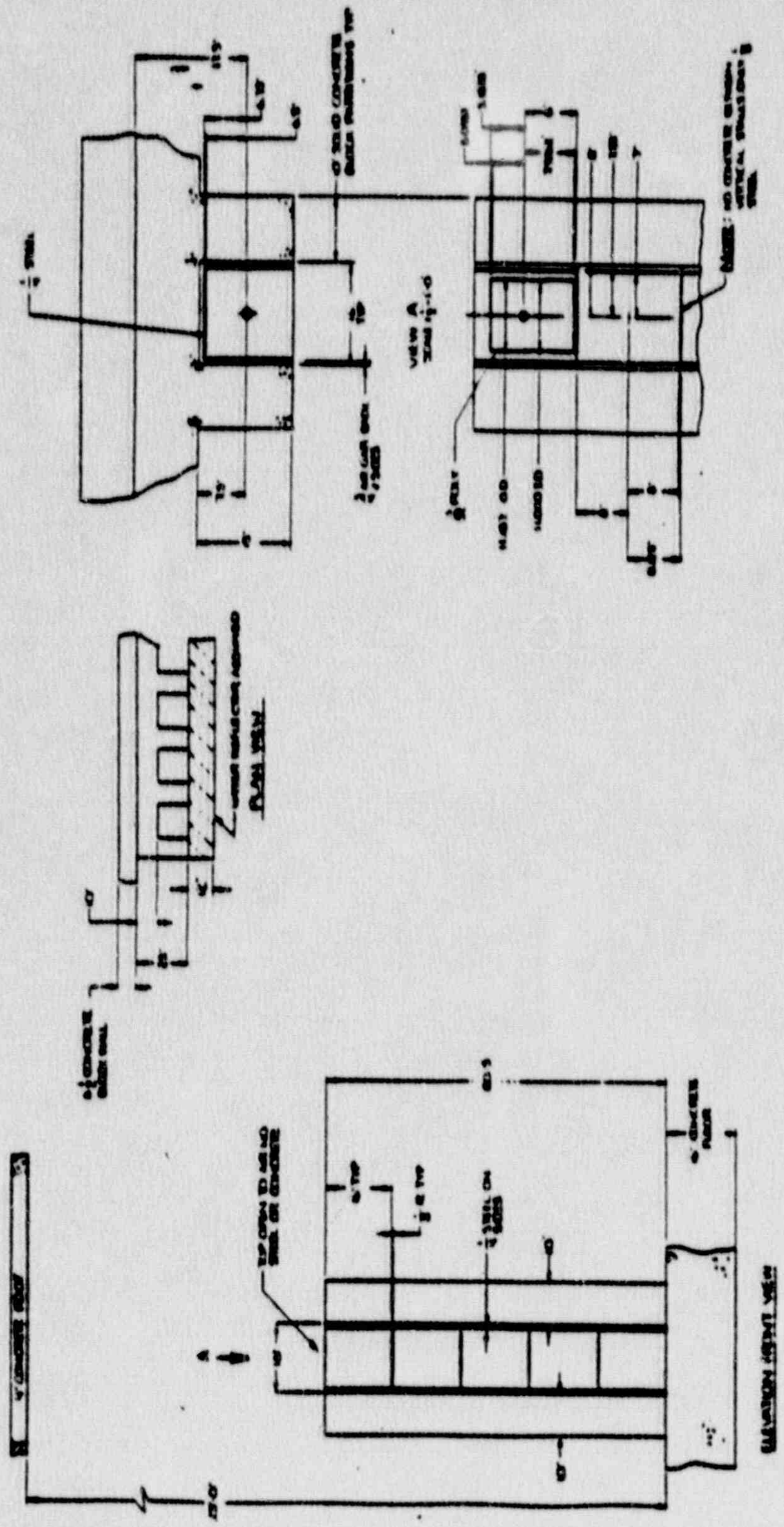

Figure 15-10

Concrete Block Storage Area

CHAPTER 16 ACCIDENT ANALYSES

The Radiological Contingency Plan for the Windsor Nuclear Fuel Manufacturing Facility, approved as Amendment 35 to License SNM-1067 on March 26, 1982, is incorporated as part of this license application. The spectrum of postulated accidents evaluated for the Windsor facility is presented in Section 3 of the Radiological Contingency Plan.