Enclosure I
ML-94-006

COMBUSTION ENGINEERING, INC.

WINDSOR SPECIAL NUCLEAR MATERIAL

LICENSE NO. SNM-1067

AMENDMENT APPLICATION

FEBRURAY 1994

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WINDSOR SPECIAL NUCLEAR MATERIAL

LICENSE APPLICATION

PARTS I & II

COMBUSTION ENGINEERING, INC.
1000 Prospect Hill Road
Windsor, Connecticut 06095-0500

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PART I. LICENSE CONDITIONS

1.0 STANDARD CONDITIONS AND SPECIAL AUTHORIZATIONS

1.1 NAME

Combustion Engineering, Inc., is incorporated in the State of Delaware with a principal office at 1000 Prospect Hill Road in Windsor, CT. The location where licensed activities are conducted is at 1000 Prospect Hill Road in Windsor, CT.

1.2 LOCATION

The mailing address for all license correspondence is:

Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, CT 06095-0500
Attn: Nuclear Materials Licensing

Licensed activities involving product research and development are conducted primarily in Building 5. Additional activities may be conducted in Buildings 2, 3, 3A, 6, 16, and 18.

Decontamination and decommissioning activities will be conducted in Buildings 17 and 21, which in the past were the principal areas used for commercial uranium-bearing fuel manufacturing and storage activities.

1.3 LICENSE NUMBER

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

1.4 POSSESSION LIMITS AND LOCATION

Combustion Engineering, Inc., requests authorization at its Windsor site for the following quantities of radioactive materials:

	Material	Form	Quantity	Location
(1)	Enriched Uranium	Any	<350 gms U-235 and <5 KG UF_6	Buildings 3, 3A, 5, 6, 16, and 18
(2)	Natural and/or Depleted Uranium	Any	10,000 KgU and <5 KG UF_6	Buildings 2, 3, 3A, 5, 6, 16, and 18
(3)	Pu-238	Sealed Neutron Sources	4 sources, each containing less than 2.0 gm Pu-238	Building 17
(4)	Uranium enriched to greater than 5.0 weight percent U-235	Residue	1000 gms U-235	Windsor Site
(5)	Uranium enriched to less than or equal to 5.0 weight percent U-235	Residual Uranium Oxides	700 gms U-235	Buildings 17 and 21

1.5 AUTHORIZED ACTIVITIES

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

Building 2 - Fuel Development Activities (natural and/or depleted uranium).

Buildings 3, 3A, 5, 16, & 18 - Fuel Development Activities.

Building 6 - Waste water processing from decontamination and fuel development activities.

Additional activities being carried out at the Windsor site involving residual contamination are as follows:

Buildings 17 & 21 -	Decontamination and decommissioning activities.
Windsor Site -	Residue from prior operations. Not to exceed 350 gms U-235 in any one location. Multiple locations to be separated from one another by a minimum of 12 feet.

1.6 EXEMPTIONS

Because of the exemptions allowed in 10CFR70.24(d), all buildings and areas shall be exempt from the requirement of a criticality monitoring system as specified in 10CFR70.24(a)(1).

2.0 ORGANIZATION AND ADMINISTRATION

2.1 ORGANIZATION RESPONSIBILITIES AND AUTHORITY FOR KEY POSITIONS IMPORTANT TO SAFETY

2.1.1 Facilities Manager

The Facilities Manager is responsible for management of the former fuel manufacturing facility (Building 17/21 Complex), now intended for decontamination and decommissioning. In this position he or she is responsible for activities within the Building 17/21 Complex that are regulated by the Nuclear Regulatory Commission. The Facility Manager's responsibilities and authority encompass the following functions: operations, accountability, security, training, criticality, radiological safety, environmental protection, transportation, materials handling and storage, licensing, equipment engineering, and maintenance.

2.1.2 Manager, Core Materials Development

The Manager, Core Materials Development is responsible and has authority for the safe conduct of laboratory operations that are regulated by the Nuclear Regulatory Commission. This responsibility includes ensuring that laboratory activities are conducted in accordance with radiological safety, criticality, security, and accountability practices.

2.1.3 Radiation Safety Officer (RSO)

The Radiation Safety Officer is responsible for defining and implementing procedures related to radiological safety. Procedures address safety criteria, monitoring, and training necessary to ensure the protection of employees, the public, and the environment. As part of this responsibility he ensures that ALARA is addressed.

The Radiation Safety Officer is responsible for supervising the radiological protection team. The radiological protection team performs radiological surveys, air sampling, and radiological safety job coverage for tasks associated with the decommissioning project and for activities conducted in the Fuel Development Laboratories. If the Radiation Safety Officer believes an operation to be unsafe, he or she has the authority to halt that operation.

2.1.4 Nuclear Criticality Specialist

The individual fulfilling this function ensures that facility operations (procedures and equipment) or changes thereto are acceptable with regard to nuclear criticality safety. The criticality specialist has the authority to halt any operation in the Building 17/21 Complex or Fuel Development Laboratories that he or she believes represents an unsafe criticality condition.

2.2 PERSONNEL EDUCATION AND EXPERIENCE REQUIREMENTS FOR KEY POSITIONS IMPORTANT TO SAFETY

2.2.1 Facilities Manager

The minimum qualifications for this position are a bachelor's degree in one of the sciences or engineering or its equivalent with five (5) years experience including at least two (2) years in management positions in the nuclear industry.

2.2.2 Manager, Core Material Development

The minimum qualifications for this position are a bachelor's degree in one of the sciences or engineering or its equivalent with two (2) years experience in the nuclear industry.

2.2.3 Radiation Safety Officer

The minimum qualifications for this position are a bachelor's degree in one of the sciences or engineering or its equivalent with two (2) years experience in health physics.

2.2.4 Nuclear Criticality Specialist

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

2.3 SAFETY COMMITTEE

Safety related Fuel Development Laboratory activities and the Building 17/21 Complex decontamination and decommissioning activities are reviewed by the Safety Committee. The Safety Committee will be comprised of, as a minimum, a representative of the Fuel Development Laboratory, a representative of the Building 17/21 Complex, and the Radiation Safety Officer. Members of the Safety Committee will be appointed by the Facilities Manager or the Manager, Core Materials Development. The Safety Committee will review abnormal occurrences and also have the responsibility to affect the ALARA program. As a minimum, the Safety Committee will meet quarterly.

A record of the proceedings of each meeting of the Safety Committee is retained for a period of at least two (2) years from the date of the meeting. A record of the closure of any Safety Committee finding is retained for two (2) years following the date of closure.

2.4 APPROVAL AUTHORITY FOR PERSONNEL SELECTION

The personnel who are in key positions important to safety and are involved in activities within the scope of this application, will be approved by the next level of management above the position to be filled. Other staff positions are filled following the prevailing administrative practices.

2.5 TRAINING

Training for personnel working with licensed material is provided commensurate with the hazards faced by the worker. The training program defines training requirements for in-house workers, contractors, and visitors.

Topics that are included in personnel training programs for persons working with licensed material include:

(1) Procedures for the storage, transfer, and use of radioactive materials.

(2) Health protection problems associated with exposure to the types of radioactive materials and radiation that are encountered.

(3) Precautions or procedures to minimize exposure (ALARA).

(4) The purposes and functions of protective devices and/or equipment employed in the Building 17/21 Complex and the Fuel Development Laboratory facilities.

(5) The need to observe, to the extent within the workers' control, the applicable provisions of Nuclear Regulatory Commission

regulations and licenses for the protection of personnel from exposures to radiation or radioactive material.

(6) The responsibility of the worker to promptly report to Combustion Engineering, Inc. management any condition which may lead to or cause a violation of Nuclear Regulatory Commission regulations and licenses or unnecessary exposure to radiation or to radioactive material.

(7) The appropriate response to warnings made in the event of an unusual occurrence or malfunction that may involve exposure to radiation or radioactive material.

(8) The radiation exposure reports which workers may request (as referenced in 10 CFR § 19.13).

2.6 OPERATING PROCEDURES

Combustion Engineering, Inc. conducts decontamination and decommissioning work using written procedures. The Radiation Safety Officer will determine if the procedures involving the storage and handling of special nuclear materials need to be reviewed and approved by the Nuclear Criticality Specialist. Routine activities in the Fuel Development Laboratory facilities involving licensed materials are conducted in accordance with written procedures. Fuel Development Laboratory activities and decontamination and decommissioning tasks may alternately be controlled by radiation work permits or task plans.

Task plans include information about both precautionary and performance related issues. Task plans emphasize ALARA as well as compliance with applicable radiological, and criticality safety requirements and help assure that radiological safety remains a high priority.

2.7 INTERNAL INSPECTIONS

Inspections are performed to determine if decontamination and decommissioning and Fuel Development Laboratory operations are being conducted in accordance with applicable license conditions and written procedures. Quarterly inspections cover radiological, criticality and environmental safety. Qualified personnel having no direct responsibility for the plant operation being inspected are used as inspectors to ensure unbiased and competent results. Items requiring corrective action are documented in a report distributed to the Facilities Manager (for the Building 17/21 Complex) or to the Manager, Core Materials Development (for the Fuel Development Laboratory facilities). Follow-up actions will be documented.

2.8 INVESTIGATIONS AND REPORTING

Abnormal occurrences are investigated and reported to the Facilities Manager or to the Manager, Core Materials Development, as appropriate. Reports to the Nuclear Regulatory Commission are made in accordance with specific conditions of this application and the applicable Federal Regulations. The level of investigation and the need for corrective action are determined based on the severity of the incident.

2.9 RECORDS

Records pertaining to health and safety, abnormal occurrences, inspections, ALARA, employee training, personnel exposures, routine radiation and contamination surveys and environmental surveys are retained to demonstrate compliance with the conditions of this application and the applicable Federal, State and local regulations. Records are retained, as a minimum, for the periods specified in the governing regulations.

3.0 RADIATION PROTECTION

3.1 SPECIAL ADMINISTRATIVE REQUIREMENTS

3.1.1 ALARA Commitment

Combustion Engineering Nuclear Operations has a strong commitment to the ALARA philosophy. As such, a key objective is to minimize exposure to radioactive material for the public, the environment, and workers at the Windsor site. The following policies are implemented for decontamination and decommissioning activities and work in the Fuel Development Laboratory facilities:

(1) ALARA targets will be set and trends will be monitored.

(2) Effluent: In the interest of limiting exposures to the public, efforts are made to reduce effluent to the minimum practicable level.

(3) Engineering Controls: The preferred method of limiting intake is the use of engineering controls. The primary engineering control is ventilation control. In cases where engineering controls are not adequate to protect the workers, respiratory protection may be used in accordance with a respiratory protection program.

The Fuel Development Laboratory facilities use engineering controls as the preferred method for controlling exposure to radioactive materials.

3.1.2 Radiation Work Permit

Work with licensed materials not covered by written procedures is covered by radiation work permits. The Radiation Safety Officer approves the issue of radiation work permits. Radiation Protection Technicians are responsible for ensuring the proper implementation of radiation work permits. Each radiation work permit is reviewed at a frequency established by the Radiation Safety Officer as part of the radiation work permit approval process and updated as appropriate.

3.2 TECHNICAL REQUIREMENTS

3.2.1 Protective Clothing

Protective clothing, when required, is prescribed by the applicable radiation work permit, task plan, or procedure.

3.2.2 Personnel Monitoring Requirements

Personnel exiting contaminated areas are required to survey themselves after removing protective clothing to ensure that they are free of contamination. Emergency evacuations are an exception to the personnel survey requirement.

3.2.3 Ventilation Requirements

Ventilation systems are designed and maintained to limit the spread of contamination into the environment by drawing air from the contaminated areas and exhausting it through HEPA filtration to the outside. This process maintains lower air pressure in the contaminated areas than in surrounding areas. Representative stack sampling is performed continuously when the HEPA ventilation systems are in operation. Samples are collected and analyzed at a minimum of once per week during ventilation system operation.

Hoods and/or other enclosures where unclad licensed material is handled are maintained during operations with a minimum face velocity of 100 linear feet per minute (100 LFPM) based on an average of multiple readings at the face of the enclosure. Face velocity measurements are obtained monthly when the hoods are available for use.

Ventilation systems which contain HEPA filters are equipped with continuous pressure differential gauges. The maximum differential pressure permitted across the HEPA filter is four inches (4") of water. The pressure drop is verified monthly when systems are in operation.

3.2.4 Instrumentation

Instruments used for radiation detection and measurement have capabilities as follows (more than one instrument may be utilized to cover the specified range):

Alpha/Beta Counting Systems: 10 DPM to 1×10^5 DPM (disintegrations per minute)

Alpha Survey Meters: 10 CPM to 1×10^5 CPM (counts per minute)

Beta/Gamma Survey Meters: 0.2 mRad/h to 500 mRad/h (millirad per hour)

Gamma Survey Meters: 0.001 mR/h to 5 mR/h (millirem per hour)

3.2.5 External Exposure (Dosimetry Requirements)

Dosimeters capable of detecting and measuring beta, gamma, and x-ray radiation are supplied to personnel who are likely to receive in one year an exposure from sources external to the body in excess of the limits specified below:

(1) Total effective dose equivalent to 0.5 rem.

(2) The sum of the deep dose equivalent and the committed effective dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 5 rem.

(3) An eye dose equivalent of 1.5 rem.

(4) A shallow dose equivalent of 5 rem to the skin or to any extremity.

An investigation of personnel radiation exposures is conducted if dosimetry results exceed fifty percent (50%) of the above limits.

3.2.6 Internal Dose Assessment

Individuals who in one year are likely to receive an intake in excess of 10 percent of the Annual Limit on Intake (ALI) are monitored for exposure to radioactive material. For individuals likely to receive an intake in excess of 10 percent of the ALI, internal dose is assessed by taking suitable and timely measurements of:

(1) Concentrations of radioactive materials in the air in the work area; or

(2) Quantities of radionuclides in the body; or

(3) Quantities of radionuclides excreted from the body: or

(4) Combination of the measurements specified.

In accordance with the provisions of 10 CFR 20.2106, records of estimated intake or body burden of radionuclides will be retained, as a minimum, until the license has been terminated.

3.2.7 Contamination Surveys

Contamination surveys are performed on a routine basis to monitor radioactive contamination. Routine contamination surveys are performed at a minimum of once per month in the former pellet shop and the Fuel Development Laboratory where work involving unclad radioactive materials is in progress. Routine contamination surveys are performed in the balance of the Fuel Development Laboratory in Building 5 and the Building 17/21 Complex once per calendar quarter. Surveys conducted in support of work performed under a radiation work permit may be used to meet the monthly survey requirement if they conform to what is required in the contamination survey procedure.

3.2.8 Respiratory Protection

The inhalation of airborne radioactive material is controlled under most circumstances by the application of engineering controls, including containment, and ventilation equipment. When such controls are not feasible or cannot be applied, respiratory protection is used. When it becomes necessary for individuals to work in areas where airborne radioactive contamination is likely to exceed the levels given in 10CFR20, Appendix B, Table I, Column 3, or for emergency situations, respiratory protection equipment is used pursuant to 10CFR 20.1703. Only respiratory protection equipment approved by NIOSH/MSHA is used.

4.0 NUCLEAR CRITICALITY SAFETY

4.1 ADMINISTRATIVE CONDITIONS

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

Administrative conditions define:

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

(2) 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 handling of SNM. Details on the technical bases and criteria employed in criticality evaluations are provided.

4.1.1 Basic Assumptions

(1) The amount of SNM present in Fuel Development, i.e., Buildings 3, 3A, 5, 6, 16, and 18 shall contain less than 350 grams of U-235.

(2) The amount of Uranium enriched to a maximum of 5.0 weight percent U-235 present within the Building 17/21 Complex contain 700 grams or less U-235.

4.1.2 Documenting Criticality Evaluations and Reviews

Criticality evaluations associated with facility changes affecting the handling and storage of SNM will be documented by the Nuclear Criticality Specialist.

Criticality evaluations will include assumptions affecting criticality safety limits and controls. If explicit analyses using validated methodologies are employed, the margin to criticality and a clear definition of assumed off-nominal conditions will be provided.

The criticality evaluations will consider potential scenarios which could lead to criticality and barriers erected against criticality in establishing applicable criticality limits and controls.

These limits and controls will be incorporated into applicable written procedures and postings and approved by the Nuclear Criticality Specialist and the Radiation Safety Officer. Day-to-day monitoring of workers for conformance to criticality limits and controls and administrative procedures is carried out by the Line Supervision and the Radiation Protection Technicians.

4.1.3 Posting of Limits and Controls

Work and storage areas where SNM is handled, processed, or stored will be posted with the nuclear safety limits and controls applicable to each area and approved by the Radiation Safety Officer and Nuclear Criticality Specialist. The Radiation Safety Officer will 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.

4.1.4 Labeling of Special Nuclear Material

Mass-limited containers employed in the handling or storage of special nuclear material will be labeled as to their contents. If SNM is in the container, the net weight, enrichment, and approximate material type will be indicated; if empty, the container will be so labeled or placed in designated areas for empty containers. Uncovered empty containers do not require an empty sign; these containers will not be intermixed with loaded containers unless all containers are located within designated storage locations.

4.1.5 Preoperational Testing and Inspection

The Radiation Safety Officer and Criticality Specialist may require, prior to startup of a new or modified process, that an inspection of equipment, procedures, and postings be carried out by the Radiation Safety Officer and the Criticality Specialist to assure completeness and consistency between safety evaluations, equipment design, installation, written procedures, and postings. This inspection will be documented as part of the records for this facility change and retained according to internal procedures.

A modified process is defined as one involving a change in equipment design, SNM amount and/or configuration, or process controls which invalidates any aspect of the previous safety analysis.

4.2 TECHNICAL CRITERIA

4.2.1 Individual Units

4.2.1.1 Safe Individual Units (SIU)

Minimum critical values of safety parameters will be based on either calculated or experimental data under conditions of optimum moderation and full reflection. To arrive at a SIU, these minimum critical values deduced from experimental data will 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 Other Subcritical Units

Other subcritical units are defined for the purpose of safe storage of SNM in multi-element arrays using the surface density method.

4.2.1.3 Criteria

(1) The possibility of accumulation of fissile materials in not readily available locations will be minimized through equipment design or administrative controls.

(2) Nuclear safety evaluations will include credible sources of internal moderation.

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

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

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

(6) Computer analysis methods will be validated in accordance with the criteria of Section 4.2.3 and Regulatory Guide 3.4, Revision 2, dated March 1986, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities." The highest effective multiplication factor derived by the validated analytical methods for credible operating conditions will be less than or equal to 0.95 including applicable biases and calculational uncertainties.

(7) The analytical method(s) used for the safety evaluation of SIUs and the source of validation of the methods will be specified.

4.2.2 Multiple Units and Arrays

Criticality safety of the less complex SNM storage operations may be based on the use of limiting parameters which are applied to simple geometries. This approach employs safe units which assume optimum moderation and full reflection. Safe units may be arrayed using the surface density method.

4.2.2.1 Spacing of Safe Units

The following criteria will be employed:

(1) Application of the surface density method of spacing safe mass, volume, or cylinder diameter limited units containing UO_2 having a U-235 enrichment of less than 5 weight percent requires meeting the following criteria:

(a) Safe mass, volume, slab, or cylinder diameter limited units will not exceed the maximum values defined in the following tabulation.

	<u>Homogeneous</u>	<u>Heterogeneous</u>
MASS (Kg UO_2)	16	16
VOLUME (L)	22	17
CYLINDER DIAMETER (in)	9.2	8.4
SLAB THICKNESS (in)	4.0	3.75

(b) The spacing areas for the safe mass, volume, or cylinder diameter limited units of (a), above, will employ spacing areas no less than those defined in the following tabulation. The minimum spacing between individual units will be no less than twelve inches. Coplanar slabs require no additional spacing; non-coplanar slabs require a minimum spacing of twelve inches.

Spacing Area(ft^2)	
MASS	3.5
VOLUME	9.0
CYLINDER	5.0 (per foot of height)

(c) Each safe unit will be centered in its respective spacing area.

(3) Nuclear safety will be independent of the degree of moderation between units up to the maximum credible mist density and at full density water. The maximum mist density will be determined by studying all 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.

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

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

(6) For purposes of the Decommissioning and Decontamination Program, all heterogeneous SNM will be considered to have a particle size of 0.050 inches for criticality safety evaluations at optimum moderation conditions.

(7) Vessels and other items of equipment requiring exclusion areas will have the limits of these areas clearly marked on the

floor. Safe units in transit will not be permitted to enter an exclusion area.

(8) The analytical method(s) used for the safety evaluation of the spacing of safe units and the source of validation of the methods will be specified.

(9) Nuclear criticality safety margins will include consideration of credible accident conditions consistent with the double contingency criterion. The highest effective multiplication factor, under normal credible operating conditions, will be less than 0.95 including a two-sigma statistical calculational uncertainty, where appropriate, as well as any other applicable uncertainties and biases.

4.2.3 Validation of Calculational Methods

Criticality safety evaluations for SNM process/storage systems requiring the use of computerized methodologies such as transport and Monte Carlo codes will employ validated models. These models will be validated by analysis of pertinent critical or subcritical experiments to define the range of applicability of the model and associated bias in calculated eigenvalues. The validation analyses for each model will be documented, consistent with ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors", independently reviewed, and retained on file for the lifetime of the use of results of that model.

The following calculational models have been validated for criticality safety analyses. Pertinent results from the validation studies are summarized below.

Sixteen Broad Neutron Group Heterogeneous Model-This model is employed for heterogeneous systems consisting of SNM/moderator/structural materials and is characterized by use of a KENO type spatial calculation. Sixteen group cross sections for these calculations are derived using the NITAWL and XSDRNPM codes and the 123 group GAM/THERMOS master library. The validation of this model is based on the analysis of experiments involving air, low density hydrogenous materials, and water interposed between four 18 x 18 clusters of 4.742 w/o UO_2 rods (J.C. Manaranche, et al, "Dissolution and Storage Experiment with 4.75 w/o U-235 Enriched UO_2 Rods", Nuclear Technology, Vol. 50, pg. 148, September 1980). A total of nine experiments were analyzed. The hydrogen densities in the cross shaped region between the four clusters vary from 0.0 to 0.0414 g/cc for the low hydrogen density materials plus water at 0.1119 g/cc. A primary objective of these analyses was to assess the accuracy of the model for interactive analyses of moderated units of SNM with interposed low density moderator. For the seven low hydrogen density and air lattices, the mean multiplication factor was 1.00449, the standard deviation is 0.00643, the 95/95 multiplier is 3.34, and the 95/95 confidence limits are 0.022. Thus, the lower critical tolerance limit is 1.00449 - 0.022 or 0.98249.

Sixteen Broad Neutron Group Homogeneous Model- This model is employed for homogeneous SNM/moderator systems and employs the Hansen-Roach sixteen group library. The validation of this model is based on using KENO-IV and the Hansen-Roach library in a recalculation of the forty experiments originally analyzed in Y-1948, (G.R. Handley and C.M. Hopper, Validation of the KENO Code for Nuclear Criticality Safety Calculations of Moderated, Low-Enriched Uranium Systems). A bias curve, defined as one minus the fitted expression to the calculated multiplication factors versus the average energy group causing fissions, ΔE, is as follows.

$$\text{Bias} = -2.19284 + 0.283367\,(\Delta E) - 0.00913413\,(\Delta E)^2$$

The methodology standard deviation in K_{eff} is calculated to be 0.00682. At a 95/95 confidence level, the multiplier on the standard deviation is 2.125. These experiments cover a range in H/U-235 between 133.4 to 971.7. In general, as the magnitude of H/U-235 decreases, the magnitude of ΔE decreases and, for values of ΔE less than -14.8, the fitted value of K_{eff} exceeds unity. Thus, the calculated K_{eff}s become more conservative below the lowest value of H/U-235 in these experiments.

4.2.4 Special Controls

The following technical criteria will be employed.

(1) Whenever more than one safe individual or safe mass unit is allowed in any given hood, positive spacing fixtures will be used to assure a minimum spacing of no less than twelve inches. Carts, limited to one safe individual mass unit will measure at least 1.871 feet on a side, and will be designed to assure that the safe mass is centered.

(2) Waste containers will individually contain less than 350 grams U-235.

(3) UNC-2901 shipping containers (Certificate of Compliance No. 6294) containing SNM will be stored within the transport vehicle or inside the Building 17/21 Complex. If stored within the vehicle, said vehicle will be within the Building 17/21 Complex security fence.

(4) The maximum internal volume of the centrifuge will be 22.0 liters. Other fissile material will be separated from the centrifuge by at least one foot.

(5) The maximum internal thickness of the slab storage tank for the centrifuge system will be 4.15 inches.

(6) The centrifuge and dump tank waste contents will be isolated from the slab storage tank when the contents of the slab storage tank are diverted to Building 6. When the dump tank, centrifuge, and slab storage tank circuit is operative, the discharge line to Building 6 will be isolated.

5.0 ENVIRONMENTAL PROTECTION

5.1 EFFLUENT CONTROL SYSTEMS COMMITMENTS

5.1.1 Low-Level Radioactive Waste

Low level radioactive wastes (LLRW) will be packaged in accordance with applicable regulations and delivered to a carrier for transport to an approved waste processor and/or disposal facility. Current copies of processor or disposal facility licenses shall be maintained.

LLRW packages awaiting shipment to a processor or disposal facility may be temporarily stored for up to one year on site. LLRW packages in such temporary storage will be checked quarterly for integrity and exterior contamination. LLRW storage areas shall be appropriately posted and the LLRW shall be secured from unauthorized removal. Prior to being placed in storage, packages will be checked for exterior contamination and labeled as to enrichment and U-235 content. LLRW may also be stored for up to five years as interim storage provided that it is protected from the elements. Interim LLRW storage shall be in Building 6, in locked C-vans or trailers, and in other buildings or enclosures on site. LLRW in interim storage will be checked annually for integrity and exterior contamination. The requirements for storage do not apply to waste which is in process in accordance with approved procedures.

Records will be maintained of the contents of LLRW packages. All packages will be stored on raised platforms (e.g., built in or portable pallets) or legs and package stacking will be limited to three (3) high. When placed in storage, the packages shall be sealed in a manner which precludes casual entry (e.g., by the use of steel clips or strapping) until final sealing is accomplished prior to shipment to the processor or disposal facility. Packages containing liquid wastes shall not be stored outside, shall be segregated from solid waste packages (e.g., stacked separately) and shall be appropriately labelled.

5.1.2 Liquid Effluents

Liquid effluents are sampled prior to discharge to demonstrate that the average activity concentrations are equal to or less than $3x10^{-7}$ μci/ml gross alpha and beta activity. Release of liquid effluents is authorized by the Radiation Safety Officer.

5.1.3 Airborne Effluents

Airborne effluents are monitored in accordance with Section 3.2.3. If the radioactivity in plant gaseous effluents exceeds 18 μCi of uranium per calendar quarter, a report which identifies the cause for exceeding the limit and the corrective actions taken to reduce the release rates will be prepared. This report will be submitted to the NRC within 30 days of the end of the calendar quarter.

Upon completion of decontamination and decommissioning activities, environmental monitoring of airborne releases from the Building 17/21 Complex will cease.

5.2 ENVIRONMENTAL MONITORING PROGRAM

The effectiveness of the effluent controls specified in Section 5.1 will be verified by sampling and analysis of media collected from the site and its environs in accordance with the Environmental Monitoring Program.

Records of environmental monitoring data will be retained, as a minimum, for a period of two years.

6.0 DECOMMISSIONING FUNDING PLAN

Combustion Engineering's Decommissioning Funding Plan dated July 2, 1992 was submitted by letter ML-92-035 from J. F. Conant (CE) to J. W. N. Hickey (NRC).

7.0 FUNDAMENTAL NUCLEAR MATERIAL CONTROL PLAN (FNMCP)

Combustion Engineering will follow Part I, chapters 1.0 through 9.0 of the current NRC approved document entitled "Fundamental Nuclear Material Control Plan", dated November 17, 1987, as well as any subsequent revisions made in accordance with the provisions of 10 CFR 70.32(c).

PART II. SAFETY DEMONSTRATION

1.0 OVERVIEW OF OPERATIONS

1.1 CORPORATE INFORMATION AND FINANCIAL QUALIFICATIONS

1.1.1 Name and Address of Licensee

Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, CT 06095-0500

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

1.1.2 Name, Address and Citizenship of Principal Officers

The following is a listing of all directors and principal officers of Combustion Engineering, Inc.

Directors

R. J. Slember	R. E. Donovan
D. P. Hamilton	T. H. Powers

Officers

R. J. Slember	L. R. Lansford	J. Guarino
R. E. Donovan	R. E. Newman	R. D. Austin
D. P. Hamilton	R. R. Rohlicek	R. S. Bell, Jr.
T. H. Powers	R. S Siudek	R. M. Burt
B. F. McNamara	E. B. Lyon	A. Kaiser
R. F. Cronin	R. W. Jewell	D. E. Harris
J. B. Faucette	R. Stein	D. E. Lewis
B. S. Sohlen	P. C. Widman	T. N. Sacco
J. P. Brett	A. E. Fournier	T. P. Morgan

These individuals, with the exceptions of R. Stein and A. Kaiser, are citizens of the United States of America. While some of these individuals may also have other offices, a common address is given below:

Combustion Engineering, Inc.
900 Long Ridge Road
Stamford, CT 06904

1.1.3 Company Background

Combustion Engineering, Inc. is a wholly owned subsidiary of Asea Brown Boveri Inc., also a corporation organized under the corporate law of the state of Delaware. Asea Brown Boveri Inc. is a wholly owned subsidiary of ABB Asea Brown Boveri Ltd., a Swiss corporation which is jointly owned by Asea AB, a publicly held Swedish corporation, and BBC Brown Boveri, Ltd., a publicly held Swiss corporation.

Combustion Engineering Nuclear Operations is a division of Combustion Engineering, Inc. Nuclear Operations operates the Fuel Development Laboratory and conducts the decontamination and decommissioning of the Building 17/21 Complex.

The commercial nuclear fuel manufacturing history of the Windsor site dates from late 1968, when SNM-1067 was first issued to Combustion Engineering, Inc. Commercial activities covered by SNM-1067 have involved development and production of fuel products for the commercial nuclear industry. Initially, uranium dioxide (UO_2) fuel pellets purchased from an outside supplier were processed and loaded into fuel rods in Building 17, the manufacturing facility at the Windsor site. In 1970, a fuel pellet operation was added to the already existing fuel pellet loading and fuel assembly operation. At that time, Building 17 began to receive uranium dioxide powder, which was subsequently pressed into fuel pellets. UO_2 pellet pressing and powder handling operations continued until permanently halted in December of 1989. Upon the cessation of powder operations, a major decontamination project was undertaken in the pellet shop of Building 17, resulting in greatly reduced contamination levels in the manufacturing facility. Since that time, manufacturing activities in Building 17 have been limited to producing finished fuel assemblies from sintered UO_2 pellets manufactured at and received from another site. Uranium bearing fuel manufacturing operations ceased on September 30, 1993. Following the cessation of uranium handling operations, Combustion Engineering Inc. initiated the decontamination and decommissioning of the Building 17/21 Complex. Fuel development activities have continued in the Fuel Development Laboratory facilities.

1.1.4 Information Known to Applicant Regarding Foreign Control

Information regarding foreign ownership of Combustion Engineering, Inc., was provided to the NRC in a letter dated November 21, 1989.

1.1.5 Financial Qualifications

Combustion Engineering, Inc. is financially qualified to engage in the proposed activities of this license application.

A decommissioning funding plan for the Windsor site areas used for commercial production activities under SNM-1067 was submitted to the NRC on July 2, 1992, under a cover letter entitled, "Decommissioning Funding Plan", from John F. Conant to John W. N. Hickey.

1.2 OPERATING OBJECTIVE

The objectives of the Combustion Engineering Building 17/21 Complex Decontamination and Decommissioning Project are as follows:

(1) Perform tasks in a safe and environmentally acceptable manner in accordance with applicable local, State, and Federal regulations.

(2) Maintain exposures to radioactive material As Low As Reasonably Achievable (ALARA).

(3) Minimize the volume of radioactive waste generated.

(4) Decontaminate the Building 17/21 Complex as required to free release the area for unrestricted use.

(5) Verify that soils inside and up to one meter outside the Building 17/21 Complex fence line, as well as the two drainage swales, are available to be free released for unrestricted use.

(6) Verify by survey that applicable criteria for release have been met.

The objectives of the Fuel Development Laboratory are as follows:

(1) Conduct development, analysis and testing of nuclear fuel and fuel-related components in a manner consistent with applicable local, State and Federal regulations.

(2) Perform chemical, mechanical and hydraulic testing of nuclear fuel, fuel assemblies, and reactor components in accordance with appropriate testing, safety, security, accountability and radiological protection procedures.

(3) Perform testing and analyses to support operational activities and programs such as decommissioning, environmental radiation monitoring, bioassay, radiological protection and waste water processing.

1.3 SITE DESCRIPTION

The Combustion Engineering Windsor site is located in north central Connecticut (Figure II-1). The site is approximately 500 acres along the section of the Farmington River known as the Rainbow Reservoir in the town of Windsor, Connecticut (Figure II-2).

1.4 LOCATIONS OF BUILDINGS ONSITE

The locations of buildings on the Windsor site are shown in Figure II-3.

1.5 HISTORY OF LICENSE

Combustion Engineering first applied for a license to process low enriched uranium in 1968. License SNM-1067 was then issued for a period of 5 years by the U.S. Atomic Energy Commission (AEC). The license has been renewed at approximately 5 year intervals since then.

Figure II-1
Combustion Engineering Location within Connecticut

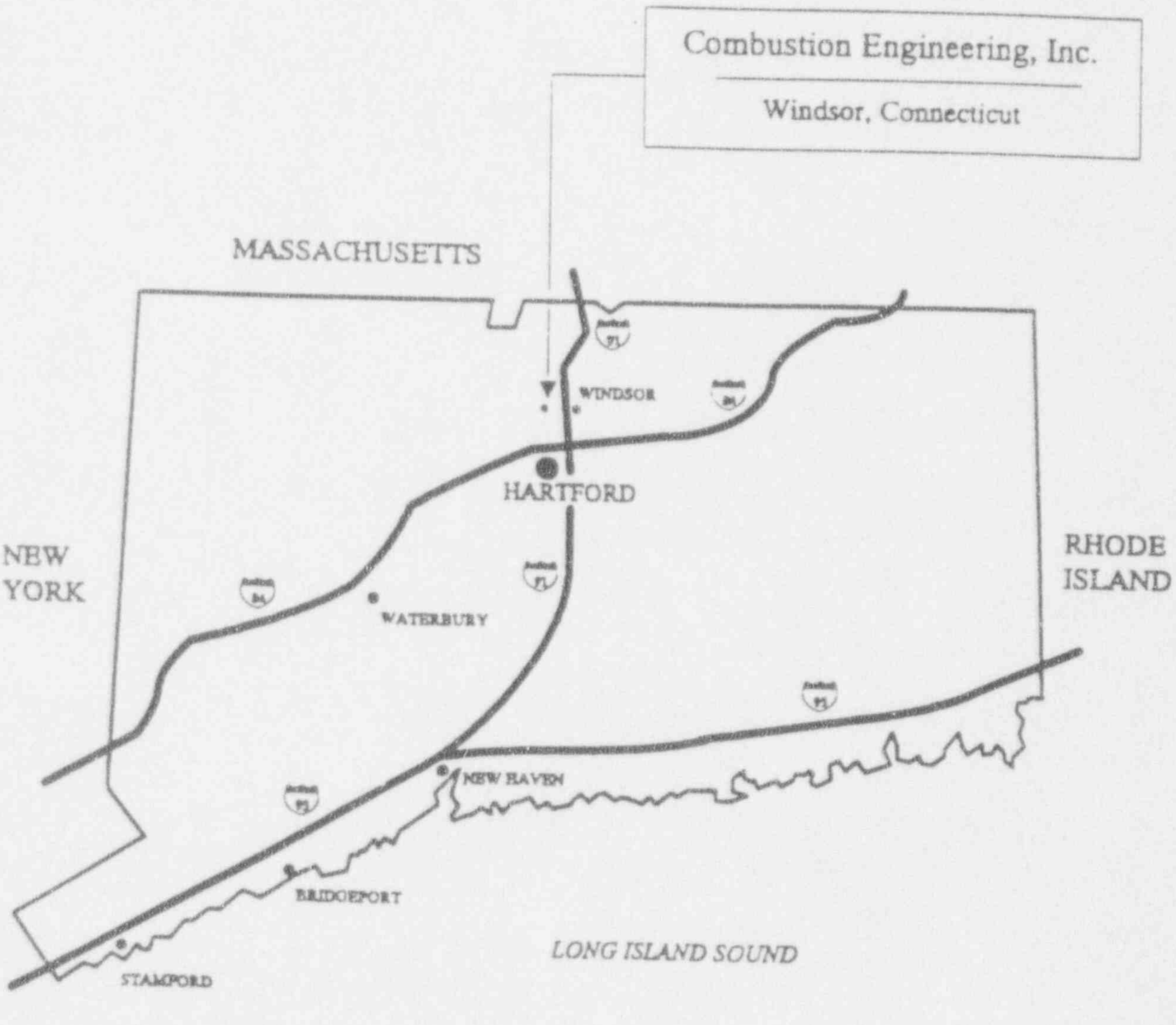


Figure II-2

Combustion Engineering Location within Local Area

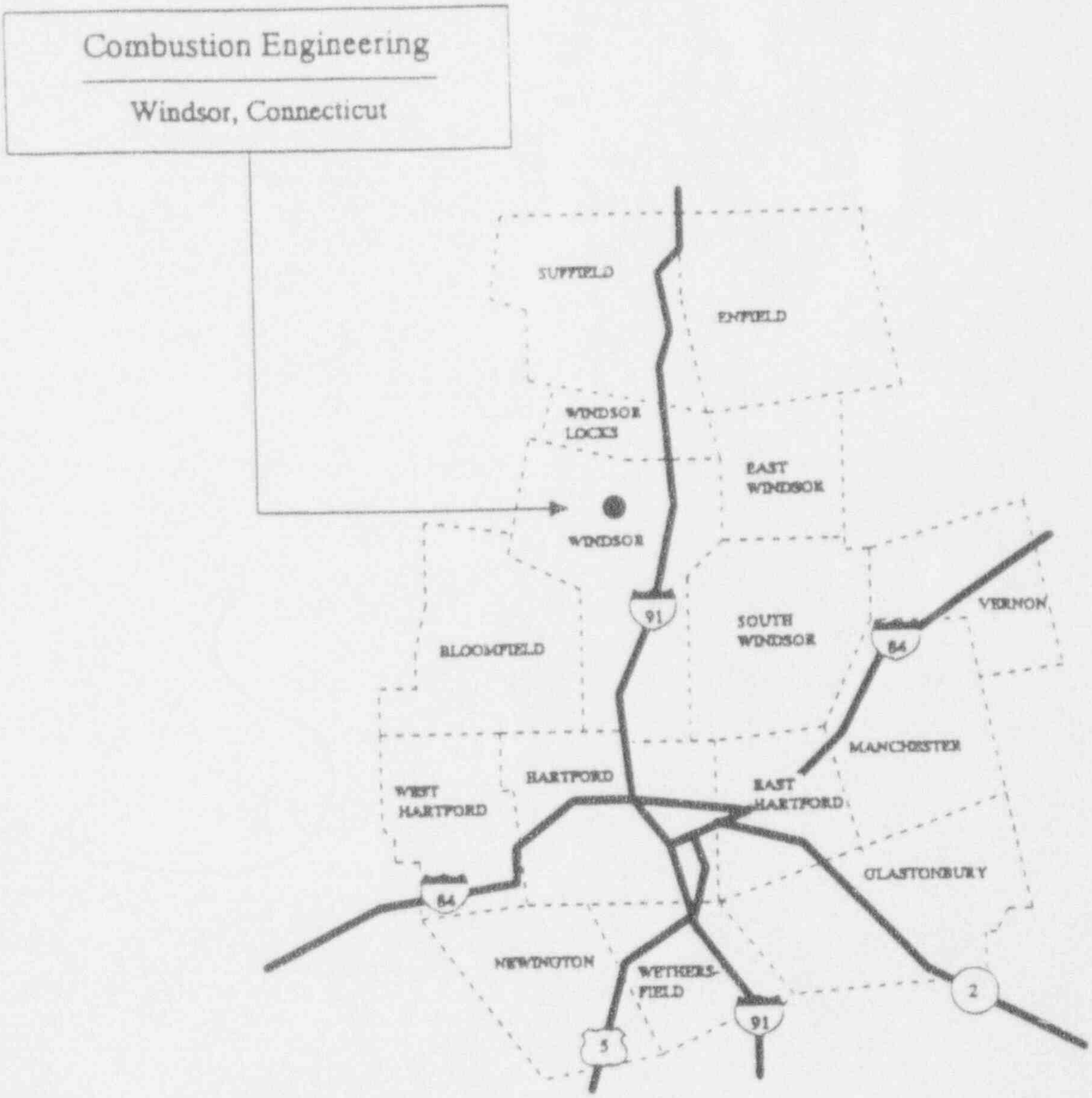
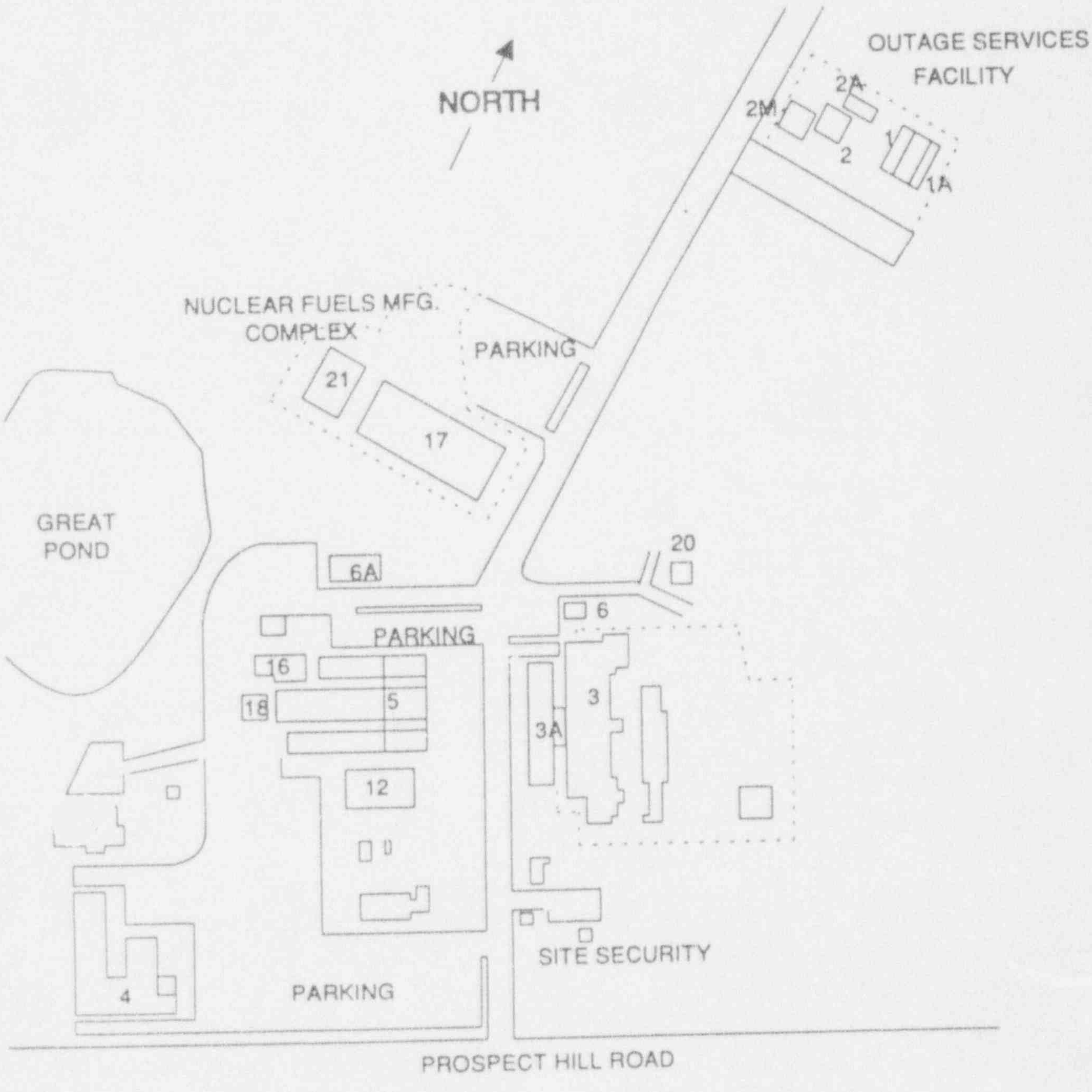


Figure II-3

Location of Buildings on the Windsor Site



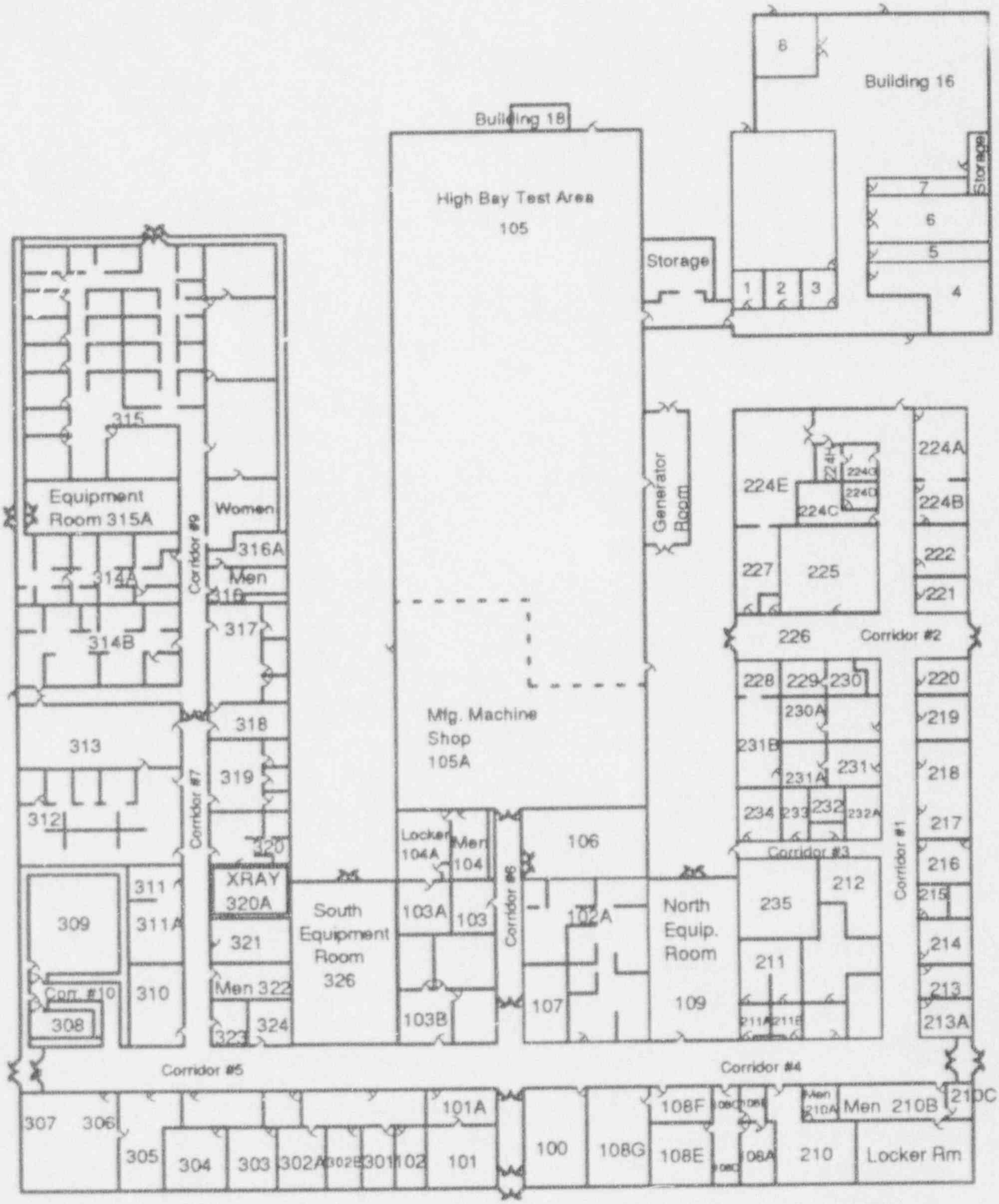
2.0 FACILITY DESCRIPTION

2.1 FUEL DEVELOPMENT LABORATORY

The Fuel Development Department maintains facilities for the development, analysis, and testing of nuclear fuel and fuel-related components of commercial nuclear reactors. These activities, which include chemical, mechanical, and hydraulic testing of uranium fuel, fuel assemblies, and reactor core components, are conducted primarily in Buildings 2, 5, and 18. The majority of these operations are conducted in Building 5. Mechanical and hydraulic testing of loaded fuel rods and assemblies containing depleted uranium is performed in Buildings 2 and 18.

Building 5, shown in Figure II.4, contains the Analytical Chemistry Laboratories and Pellet Physical Testing Laboratory. The Analytical Chemistry Laboratories consist of a Uranium Analysis Laboratory, Environmental/Bioassay Laboratory, Radiochemistry Laboratory, Environmental Laboratory, and a Radiochemistry Counting Room. Chemical analysis of uranium fuel and various reactor core materials and components is conducted in the Analytical Chemistry Laboratories. These laboratories also perform analyses which support operational activities such as environmental radiation monitoring, personnel bioassay, radiological protection, and waste water processing. The Pellet Physical Testing Laboratory is used for activities such as pellet resintering tests and immersion density tests.

Figure II-4 Building 5



2.2 BUILDING 17/21 COMPLEX

Commencement of uranium handling operations in Building 17 coincided with the issuance of SNM-1067 late in 1968. The facility was divided into a "clean" area (with no radioactive contamination hazards) for manufacture of fuel assembly hardware and a "hot" area (with complete radioactive contamination controls in place) for uranium fabrication work. Building 21 was operated as a clean facility, serving as a materials warehouse for production supplies and packaged radioactive materials. The land areas inside the fence were maintained clean and utilized in support of shipping and receiving activities. The material involved in fuel production was then, and remains today, limited to uranium dioxide enriched in the U-235 isotope to a maximum of five percent by weight. Commercial operations have involved receipt of UO_2 in pellet or powder form, pressing of the powder into pellets, and loading of the pellets into fuel rods that were assembled into fuel bundles. General records indicate that minor spills which occurred occasionally in the pellet shop were generally associated with production operations, and were promptly cleaned up. There are no records of spills or accidents with the potential to impact the health and safety aspects of the decommissioning project. In December, 1989, operations with UO_2 powder ceased permanently and an extensive decontamination project was undertaken. As a result of this project, UO_2 contamination levels in the shop were dramatically reduced. In its current configuration, the pellet shop is only minimally contaminated with UO_2.

3.0 ORGANIZATION AND PERSONNEL

The Windsor Decontamination and Decommissioning and Fuel Development organization structures are depicted in Figure II-5.

3.1 FUNCTIONS OF KEY PERSONNEL

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

3.1.1 Supervisor, Analytical Chemistry

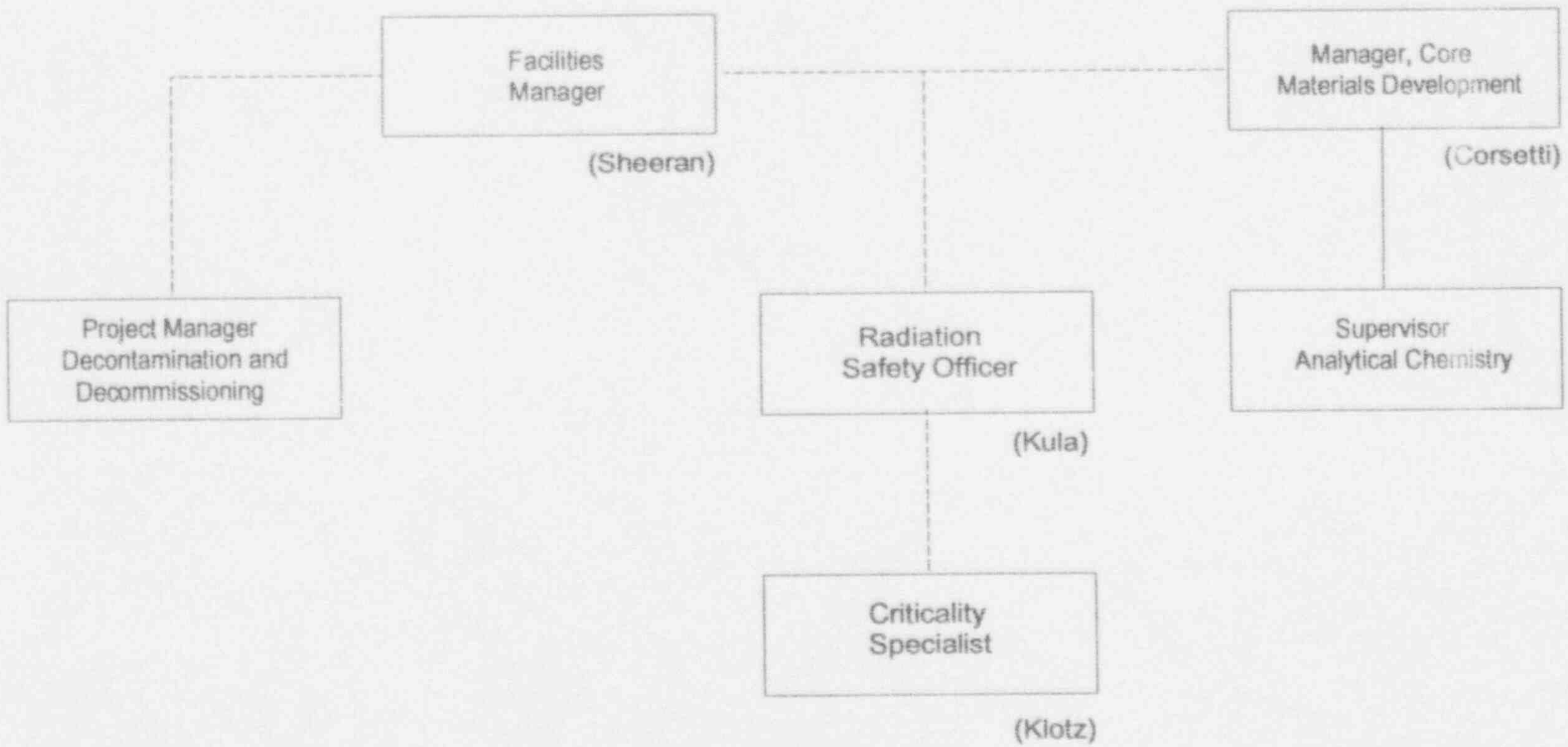
The Supervisor, Analytical Chemistry is responsible for direction of Analytical Chemistry laboratory operations. This responsibility includes overseeing laboratory materials handling, storage, and disposal practices. The Supervisor is also responsible for ensuring that laboratory measurements are accurate and traceable to accepted industry standards.

3.1.2 Project Manager, Decontamination and Decommissioning

The Project Manager, Decontamination and Decommissioning is responsible for overseeing the planning and execution of day to day decontamination activities for the Building 17/21 Complex.

The Project Manager supervises the decontamination team. This team performs most of the decontamination work associated with decommissioning of the Building 17/21 Complex. This includes cleaning structures and materials and dismantling structures and equipment as required.

Figure II-5: Decontamination and Decommissioning
and Fuel Development Laboratory Organizations



* 3.2 RESUMES OF KEY PERSONNEL IMPORTANT TO SAFETY

Resumes of key personnel important to safety are provided on Pages II.3-4 through II.3-12.

ROBERT E. SHEERAN - Manager, Facilities

EDUCATION

Bachelor Mechanical Engineering - New York University
Masters Mechanical Engineering - New York University
Masters Business Administration - University of Hartford (36 of 48 credit hours)

PROFESSIONAL EXPERIENCE

COMBUSTION ENGINEERING, INC.
Manager, Facilities 1993 - Present

Responsible for NRC regulatory compliance for the former nuclear fuel manufacturing facility, which is now intended for decontamination and decommissioning.

Manager, Accountability and Security 1988 - 1993

Responsible for establishing and maintaining a security program to comply with NRC regulations. Responsible for establishing and maintaining an Accountability program to comply with NRC regulations. Supervised and directed Security and Accountability personnel in support of the Manufacturing operation. Responsible for the preparation/implementation/maintenance of Security and Accountability manuals that define the manner in which the Security/Accountability programs are carried out.

Manager, Nuclear Licensing, Safety, Accountability & Security 1983 - 1988

Total responsibility for compliance with DOT, NRC, and OSHA regulations at Combustion Engineering's uranium Fuel Fabrication Facility. Responsible for all licensing submittals and management of the following programs:

Nuclear Criticality Safety
Health Physics

Security and Security Programs
Industrial Safety
Packaging and Transportation of Radioactive Material
Nuclear material accountability of $100 million uranium inventory and associated computer system
Emergency planning, including interface with State of Connecticut Office of Civil Preparedness
Responsible for audit of CE's oxide conversion facility in Hematite, Missouri

Supervisor, Health Physics & Safety; Supervisor, Security 1981 - 1983

Responsible for compliance with DOT, NRC, and OSHA regulations at CE's uranium Fuel Fabrication Facility. Responsible for all licensing submittals and management of the following programs:

Nuclear Criticality Safety Compliance
Health Physics
Industrial Safety
Packaging and Transportation of Radioactive Material
Supervision of Health Physics Technicians
Supervision of Facility Security Force and Implementation of Security Program
Responsible for audit of Research/Development Labs in Windsor, Connecticut

Supervisor, Special Projects; Supervisor, Security 1979 - 1981

Served in the dual capacity of Special Projects Supervisor and Supervisor of Security. Special Projects Supervisor duties: long range planning for equipment and facilities, proposal development for new facilities, preparation of capital appropriation requests for new equipment and facilities to meet projected production requirements, coordinating A/E and regulatory agency requirements. Maintaining up-to-date awareness of Federal, State, and local requirements with regard to environmental impact of new equipment and facilities. Supervisor, Security duties: supervise facility security force, prepare licensing submittals for NRC approvals, perform other duties as assigned and as directed by the General Manager.

Supervisor, Nuclear Licensing and Safety 1976 - 1978

Duties and responsibilities as noted above.

RICHARD S. KULA - Radiation Safety Officer

EDUCATION

University of Connecticut, Storrs, CT, B.S., Animal Sciences, 1977.
Auburn University, Auburn, AL, Postgraduate Study in Animal Sciences

EXPERIENCE

COMBUSTION ENGINEERING, INC.

Radiation Safety Officer — Present

Define and implement radiological safety procedures and ALARA Plan for the Windsor Commercial Nuclear Fuel Manufacturing decommissioning project and Fuel Development laboratories. Supervise the radiological protection team in their performance of surveys, air sampling, radiological or industrial safety job coverage for tasks associated with decommissioning and product development.

Supervisor, Radiological Protection and Industrial Safety — 1991 - 1993

Responsible for supervision of the Nuclear Fuel Manufacturing and Fuel Development radiological protection and industrial safety program. Directed surveillance of manufacturing and laboratory activities related to radiological and industrial safety, and ensured that operations were conducted in accordance with Federal, State and local regulations. Responsible for the training and supervision of the Radiation Protection Technicians assigned to these areas.

Senior Radiological Protection and Industrial Safety Technician — 1989 - 1991

Initially assigned the third shift Lead Technician position, oversaw technicians having the responsibility of controlling the use and handling of nuclear fuel through its processing from uranium dioxide powder into a pelletized form, and subsequent fabrication into commercial reactor fuel assemblies. Later assigned Lead Technician for the decommissioning of this plant oxide handling and pelletizing equipment, as well as preparations for facility decommissioning.

INTERSTATE NUCLEAR SERVICES

Assistant Plant Manager 1987 - 1989

Responsible for all second shift plant operations at this nuclear decontamination laundry, servicing power plants, pharmaceutical companies and others. Served as Radiation Safety Officer for the second shift. Oversaw the receiving and shipping of radioactive materials and the generation of documents required by state and U.S. D.O.T. for the shipping of these materials.

COMBUSTION ENGINEERING, INC

Health Physics Technician 1981 - 1987

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

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

PROFESSIONAL EXPERIENCE

COMBUSTION ENGINEERING, INC.

Principal 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. As Nuclear Criticality Specialist, provides technical support and criticality audit function at both the Windsor Manufacturing and the Hematite Fuel Manufacturing facilities. Involved in solving special physics problems.

Section Manager, Radiation and Criticality Physics — 1965 - 1977

Responsible for radiation shielding, the ex-core criticality, and determination of source terms for Nuclear Steam Supply Systems. Also responsible 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 (continued)

CONVAIR DIVISION OF GENERAL DYNAMICS
<u>Physicist</u> 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.

LAWRENCE V. CORSETTI - Manager, Core Materials Development

EDUCATION

B.S., Materials and Metallurgical Engineering, University of Pittsburgh, 1968

M.S., Management Science, Rensselaer Polytechnic Institute, 1973

M.S., Materials, Rensselaer Polytechnic Institute, 1974

PROFESSIONAL EXPERIENCE

COMBUSTION ENGINEERING, INC.

Manager, Core Materials Development, Nuclear Fuel — 1989 to present

Responsible for line management of the Core Materials Development section, which includes irradiation testing, materials development, and analytical chemistry functions. Principle responsibility for planning and implementation of development programs related to fuel, cladding and control material testing, and performance evaluation. Participates in several joint cooperative R&D efforts in the fuel area and is responsible for monitoring of the performance reliability of CE supplied fuel.

Supervisor, New Product Design — 1985 to 1989

Supervised design group responsible for activities associated with the development of CE and non-CE reload fuel and other new products, such as control rods and burnable poison assemblies. Mechanical design related software and performance codes also were developed.

Supervisor of Irradiation Testing — 1973 to 1985

Supervised irradiation test group responsible for planning, executing, and evaluating fuel irradiation experiments and site fuel inspection programs. Program Manager for a long-term fuel performance program jointly sponsored with EPRI. Also responsible for the design and development of techniques for the non-destructive examination of fuel rods, assemblies, and control elements at reactor-site spent fuel pools.

LAWRENCE V. CORSETTI (continued)

GENERAL ELECTRIC, KNOLLS ATOMIC POWER LABORATORY
Irradiation Test Engineer

Responsible for planning, executing, and evaluating in-pile and post-irradiation nuclear reactor fuel and control material testing programs. This included directing core component examination programs to investigate fuel assembly performance.

WESTINGHOUSE NUCLEAR FUELS DIVISION
Associate Engineer

Obtained extensive experience with fuel and cladding material testing and evaluation practices. Developed and implemented standardized procedures for conducting out-of-pile cladding corrosion testing program as well as evaluating test results to determine material performance.

PUBLICATIONS

Numerous publications on various subjects related to fuel and control element performance, irradiation tests, and poolside fuel inspection techniques.

4.0 RADIATION PROTECTION PROCEDURES AND EQUIPMENT

4.1 WRITTEN PROCEDURES

Operations in the Building 17/21 Complex involving licensed material are conducted in accordance with written procedures, task plans, and/or radiation work permits. Procedures and task plans for the Building 17/21 Complex are approved by the Facilities Manager. Radiation work permits are approved by the Radiation Safety Officer

Operations in the Fuel Development Laboratory involving licensed materials are conducted in accordance with written procedures and/or radiation work permits.

4.2 INSTRUMENTS

Alpha/Beta counting systems are used in the Building 17/21 Complex and the Fuel Development Laboratory for radiation detection and measurement. The Alpha/Beta counting systems are calibrated daily through background and efficiency determinations.

The Fuel Development Laboratory is also equipped with gamma spectrometer systems for radiation detection and measurement. Operational integrity of the gamma spectrometer systems is verified by weekly gamma energy and efficiency calibration checks.

Radiation sources used in the instrument calibrations are certified to comply with Regulatory Guide 4.15 requirements for achieving measurement traceability.

5.0 OCCUPATIONAL RADIATION EXPOSURES

With cessation of the uranium processing operations the potential for a release of radioactive material is greatly diminished. Combustion Engineering, with its long standing commitment to ALARA, will continue its emphasis on exposure control to minimize intake of uranium. The ALARA indicators used include the following:

(1) Liquid Effluent: A measure of the amount of uranium released in liquid effluent.

(2) Air Effluent: A measure of the amount of uranium emitted from the Building 17/21 Complex and the Fuel Development Laboratory via the ventilation system.

(3) Shallow Dose Equivalent: Shallow Dose Equivalent is the external dose to the skin.

(4) Total Effective Dose Equivalent (TEDE): TEDE is the sum of the deep dose equivalent and committed effective dose equivalent.

(5) Airborne Radioactivity: A measure of the concentration of radioactivity in the ambient work place air. It is measured through the use of air sampling equipment and expressed in units of μCi/ml (Bq/m^3).

(6) Contamination: This is a measure of the amount of uranium surface contamination in the work environment, expressed in units of dpm/100cm^2.

6.0 ENVIRONMENTAL SAFETY

6.1 LIQUID EFFLUENTS

Liquid effluents from Fuel Development Laboratory and Decontamination and Decommissioning activities are transferred to any one of ten 2,000 gallon retention tanks, located in Building 6. Sinks and showers in the laboratories and Building 17 are also drained to these retention tanks, providing additional dilution. Before these tanks are discharged to the site creek, which flows into the Farmington River, a sample is withdrawn and forwarded to the Radiochemistry Laboratory for gross alpha and beta analyses. Release of liquid effluents are authorized by the Radiation Safety officer.

6.2 AIR EFFLUENTS

Air effluents may be released from Building 17 as a result of the decontamination and decommissioning activities. Airborne effluents are exhausted through two separate systems. Fixed Air System 1 has a capacity of 12,100 CFM and incorporates prefilters and a double bank of 12 HEPA filters. Fixed Air System 3 has a capacity of 17,500 CFM and contains prefilters and a double bank of 16 HEPA filters. Both systems are sampled continuously when the HEPA systems are in operation.

Air effluents may be released from the Fuel Development Laboratory as a result of airborne activity during handling and transfer of UO_2 for chemical analysis purposes, and metallographic examination of production fuel and special test fuel. All airborne effluents are exhausted from the Fuel Development Laboratory in Building 5 via four individual stacks. Stacks used for the exhausting of radioactive effluents are equipped with sampling connections. The exhaust is continuously sampled whenever operations involving releases of radioactive material are in progress. All but one of the stacks has absolute filters. The one exception is the environmental test laboratory stack (Stack 7).

Building 5 exhaust stacks typically have the following flows:

Stack No.	Area Monitored	Flow (ft3/min.)
2	Analytical Chemistry Lab	1700
3	Mass Spectrometry Lab	1400
5	Radiochemistry Lab	2700
7	Environmental Test Lab	2000

Air from system Nos. 2, 3, and 5 pass through single banks of HEPA filters and are vented to the atmosphere. Ventilation system filters and/or prefilters will be changed, rotated, or knocked down wherever a pressure drop of 4 inches of water is measured across the combination of the prefilter and first bank of absolute filters.

7.0 NUCLEAR CRITICALITY SAFETY

7.1 SURFACE DENSITY MODEL

The surface density model may be used to structure safe arrays of mass, volume, or cylinder diameter limited units of SNM. The generally acceptable surface density model of Regulatory Guide 3.52 (Rev. 1, Nov. 1986) is one wherein each individual unit has a fraction critical of ≤ 0.3 and the spacing of individual units within the array is such that the smeared surface density of fissionable material is no greater than 25 percent of the critical surface density of a water reflected slab of the same composition. A comparison of these limits to, for example, Figure 26 of Reference 1 indicates that the above model is conservative. However, for certain applications, it may be overly restrictive and not allow the use of unit sizes approaching those of a SIU at the possible expense of a smaller surface density ratio. If one examines the data of Figure 1 of Reference 2, there is evidence to suggest that larger values of the surface density ratio may be permitted in arrays having lower limits of the maximum achievable U-235 concentration, as in low enrichment processing facilities.

The surface density model implemented by this license is not defined by specific limits on fraction critical and ratio of smeared to critical surface density. Instead, limits are placed on the operational parameters such as mass, volume, cylinder diameter, and spacing area. The spacing area limits are independent of enrichment. One consequence of the latter choice is that both fraction critical and ratio of smeared to critical surface density may exhibit an enrichment dependence.

The concept of fraction critical is based on somewhat arbitrary definition deduced from correlations of experimental data; see, for example, Reference 3. 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 safe units for this license, non-spherical units are reduced to spherical shapes using buckling conversions based on the following equations:

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

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

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

For convenience, unreflected and water reflected extrapolation lengths, λ, are taken from Figure 2 of Reference 3.

Unreflected	Reflected
λs = 2.1 cm	5.9 cm
λt = 2.7 cm	6.6 cm
λc = 2.25 cm	6.3 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.

Table II-1 summarizes the administrative limits on mass, volume and cylinder diameter limits versus enrichment applicable to surface density modeling by the parameters described in this section; the SIU limits are provided for comparison. Note the administrative limits incorporated into Section 4.2.2.1 of Part I are for 5 w/o enriched UO_2.

The homogeneous and heterogeneous SIU limits in the data of Table II-1 are derived from DP-1014[4]. For homogeneous UO_2, the safe mass limits are lower than the SIU values and the fraction critical (fc) for the safe mass values are no greater than 0.25. The safe volume limits are also less than the SIU limits and the fraction critical is no greater than 0.40. The safe cylinder diameter limits are also less than the SIU limits and the fraction critical does not exceed 0.32.

For heterogeneous UO_2, the safe mass limits are numerically equal to the homogeneous safe mass limits for enrichments greater than 4.1 w/o U-235. SIU limits are given for two types of heterogeneous UO_2. The lower SIU limits are based on optimum pellet diameter versus enrichment. The second (higher) SIU limits are based on a pellet diameter of 0.30 inches. The latter diameter is less than the diameter of pellets routinely manufactured and is close to the optimum pellet diameter at the lower end of the enrichment range of Table II-1. Thus, these limits are the more practical limits and permit a standardization on the numerical value of the mass limits for both homogeneous and heterogeneous UO_2 at the higher enrichments.

The fraction critical for each heterogeneous safe mass limit does not exceed 0.27. The safe volume limits for heterogeneous UO_2 are less than or equal to the SIU limits. Note that the entries in the two SIU columns are identical, thus the optimum pellet diameter is close to 0.30 inches. For the cylinder diameter limits, the fraction critical does not exceed 0.33 and again the entries in the two SIU columns are identical.

The minimum spacing areas for the safe mass, volume, and cylinder diameter limits are listed below; these minimum spacing areas are enrichment independent.

Unit Type	Spacing Area (ft^2)	
Mass	3.5	
Volume	9	
Cylinder Diameter	5	(per foot of cylinder height)

The above spacing areas result in the following ratios of smeared to critical surface density for each unit type at 5 w/o U-235.

Unit Type	$\sigma s/\sigma c$ or ts/tc	fc
Mass	0.36	0.24
Volume	0.21	0.40
Cylinder Diameter	0.22	0.32

Since either the fraction critical or the surface density ratio violates the criteria of the generally acceptable surface density model of Regulatory Guide 3.52, KENO-IV[5] analyses of reflected, infinite arrays of air reflected mass, volume, and cylinder diameter limited units were performed to verify that the arrays were indeed subcritical. The following assumptions were employed in these analyses.

(1) The unit containers are assumed to be cylindrical in shape, uncovered, and have a 0.025 inch thick carbon steel side wall and floor. Container diameters were varied from 9.5 to 11.75 inches. Cylinder diameter limited containers are assumed to be

48 inches in height; mass and volume limited containers were varied in height to yield the required volume or mass (at appropriate density).

(2) The unit containers were assumed to be reflected from below by 16 inches of concrete and from above by 12 inches of water at the top of the container.

The homogeneous UO_2 and water mixture mass limited container analysis results are graphically summarized in Figure II-6 where the KENO K_{eff}, including uncertainties and bias (see Section 7.2.1), is plotted versus uranium concentration. The lower curve is for a 17 Kg mass of UO_2 at 5 wt % U-235 in a 11.75 inch ID container. Each container has a spacing area of 3.5 ft^2. The maximum K_{eff} occurs at a uranium concentration of ~0.6 KgU/L, as expected, from the observed minimum in the tabulation of gms U-235/cm^2 for 5 w/o enriched homogeneous UO_2 in Reference 4. At a concentration of 0.6 KgU/L the unit mass container diameter was decreased from 11.75 to 10.5 to 9.5 inches in successive calculations; the multiplication factor decreased with container diameter. In these analyses, the unit mass was preserved by increasing the container height.

The second curve is for a mass unit of 41 $KgUO_2$ at an enrichment of 3 w/o U-235 in a 11.75 inch ID container. Here we note the multiplication factor peaks out at ~1.1 KgU/L which again is reasonably consistent with the concentration at which the gms U-235/cm^2 is minimized in Reference 4 for homogeneous UO_2 at this enrichment level.

From these two curves we may conclude that homogeneous safe mass limits of Table II-1 at 5 and 3 w/o U-235 yield an acceptable subcritical margin when arrayed with a spacing area of 3.5 ft^2 per unit.

Figure II-7 summarizes the results of the volume and cylinder diameter limited analyses. The curves of multiplication factor versus uranium concentration exhibit a maximum at approximately 1.8 KgU/L for 5 w/o enriched homogeneous UO_2. Once again, this is consistent with the concentration at which the minimum slab thickness occurs in Reference 4. The plots of multiplication factor versus ts/tc provide a measure of the sensitivity of the multiplication factor to changes in both ts/tc and cylinder diameter or unit volume.

The curves of Figure II-7 demonstrate the acceptability of the safe volume and cylinder diameter limits of Table II-1 for homogeneous UO_2 at 5 wt% U-235, when these units are arrayed with a spacing area of 9 ft^2 for a volume unit and 5 ft^2 per foot of height for a cylinder diameter limited unit container.

For postulated uniform mixtures of UO_2 pellets and water, it is noted, at a given enrichment, the minimum gms U-235/cm^2 and slab thickness occur at the same gms U/cc of the mixture as with uniform mixtures of homogeneous UO_2 and water. A KENO array analysis for mass units of 17 Kg UO_2 of 0.325 inch diameter, 5 w/o enriched pellets with a uranium concentration of 0.6 KgU/L yields a multiplication factor including uncertainties and bias, of 0.9091 versus 0.9066 for the comparable homogeneous UO_2 case. The only difference in the array analyses, other than the degree of heterogeneity, is the heterogeneous calculation used 16 inches of water instead of concrete. The heterogeneous UO_2 array result provides an acceptable margin of subcriticality. The fact that the multiplication factor is slightly higher than for the homogeneous case is not surprising since the fraction critical is higher for the heterogeneous case. No analyses were done for the heterogeneous volume and cylinder diameter limits since the fraction critical values are in general less for the heterogeneous limits compared to the homogeneous limits. Thus, for the same spacing areas, the multiplication factors should be comparable, or less than, those computed for the arrays of homogeneous UO_2.

The safe slab limits of Section 4.2.2.1 of Part I are derived from Reference 4 for homogeneous $U(5)O_2$ or heterogeneous $U(5)O_2$ (optimum particle size of 0.050") using the safety factors of Section 4.2.1.1 of Part I.

The safe mass, volume, and cylinder diameter limits of Table II-1, the safe slab limits of the previous paragraph, and the spacing area limits are incorporated into the tabulations of Section 4.2.2.1 of Part I of this license.

7.2 VALIDATION OF CALCULATIONAL METHODS FOR CRITICALITY SAFETY

7.2.1 Homogeneous UO_2 - Water Configuration

Homogeneous fuel-water mixtures are analyzed with the KENO-IV code. In these cases the fuel may be in powder or granular (mean particle size ≤ 130 microns) form, admixed with other elements, and assumed to be uniformly mixed with water. The fuel-water mixture may be contained in vessels within a regular or irregular array. Moderator may be assumed to exist in the space between these vessels. In these analyses, the primary library source is the 16 group Hansen-Roach cross section library (revised version dated 9/17/90).

For purposes of assessing the magnitude of a possible bias in the KENO-IV code and revised sixteen group Hansen-Roach cross sections implemented on the engineering workstations, the experiments of Reference 6 were rerun. Figure II-8 shows a second order polynomial fit to the KENO-IV results for the forty experiments. This polynomial is of the following form:

$$K_{eff}]_{fit} = 3.19284 - 0.283367\ (\Delta E) + 0.00913413\ (\Delta E)^2$$

where ΔE is the average energy of the neutrons causing fissions.

The bias between the KENO calculated K_{eff} and the second order polynomial fit to the KENO results is given by the following expression:

$$\text{bias} = K_{eff}]_{KENO} - K_{eff}]_{fit}$$

Table II-2 summarizes the evaluation of the methodology standard deviation.

The methodology standard deviation is 0.00682 ΔK and the 95/95 confidence limit multiplier is 2.125.

7.2.2 Sixteen Broad Neutron Group Model

This model is similar to that employed for homogeneous UO_2 - water mixtures. It differs in that the broad group cross sections are calculated by the NITAWL and XSDRNPM code[5] sequence to take into account the heterogeneity of the fuel moderator mixture.

Critical experiments on the interposition of low hydrogen density materials between four 18 x 18 clusters of 4.742 w/o enriched UO_2 rods were performed by the Department of Nuclear Safety of the French Atomic Energy Commission and reported in Reference 7. The fuel rods are spaced on a square pitch of 13.5 mm, contain UO_2 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 II-9 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 thicknesses 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 $(CH_2)_n$,
(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[8]. The NITAWL and XSDRNPM codes are used to generate 16 broad neutron energy group cross sections. NITAWL is used to generate self shielded 123 group cross sections from 123 group super-XSDRN library (DLC-16). The resulting working library is then collapsed into a homogenized 16 energy group library in a typical fuel pin cell environment using XSDRNPM. XSDRNPM is also used to obtain separate 16 group cross section sets for structural materials and external moderators. The KENO model employed a homogenized fuel pin representation in the interior of the fuel rod cluster. The cross shaped box, the outside moderator, tank wall, lattice grid, fuel pin lower plug, bottom plate and support plate are all explicitly represented. Table II-3 summarizes the multiplication factors computed by KENO IV for nine 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, $\bar{X}$	1.00449
Standard Deviation	0.00643
95/95 Multiplier	3.34
95/95 Confidence Limits	0.022
Bias ($\bar{X}$-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 calculations.

7.3 DATA SOURCES

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

2. R. L. Stevenson and R. H. Odegaarden, "Studies of Surface Density Spacing Criteria Using KENO Calculations", Transactions of the American Nuclear Society, Vol 12, p. 890, 1969.

3. H. C. Paxton, "Correlations of Experimental and Theoretical Critical Data; Comparative Reliability, Safety Factors for Criticality Control", LAMS-2537, Los Alamos Scientific Laboratory, 1961.

4. H. K. Clark, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water", DP-1014, Savannah River Laboratory, February 1966.

5. "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation - Book II", NUREG/CR-0200.

6. G. R. Handley and C. M. Hopper, "Validation of the KENO Code for Nuclear Criticality Safety Calculations of Moderated, Low-Enriched Uranium Systems", Y-1948, June 13, 1974.

7. J. C. Manaranche, et al., "Dissolution and Storage Experiments with 4.75 w/o U-235 Enriched UO_2 Rods", Nuclear Technology, Vol. 50, Pg. 148, September 1980.

8. W. R. Cab.e, "123 Group Neutron Cross Section Data Generated from ENDF/B-II Data for Use in the XSDRN Discrete Ordinates Spectral Averaging Code", DLC-16, Radiation Shielding Information Center, 1971.

Table II-1

MASS, VOLUME, AND CYLINDER DIAMETER LIMITS VERSUS ENRICHMENT

U-235 Enrichment w/o	Mass Limits						
	Homogeneous			Heterogeneous			
	$KgUO_2$	fc	SIU	$KgUO_2$	fc	SIU[1]	SIU[2]
>Nat. ≤ 2.5	54	.19	83	50	.23	57	58
>2.5 ≤ 3.0	41	.22	54	38	.25	41	41
>3.0 ≤ 3.2	36	.22	49	36	.26	36	37
>3.2 ≤ 3.4	35	.24	43	33	.27	32	34
>3.4 ≤ 3.6	32	.25	38	30	.27	29	30
>3.6 ≤ 3.8	28	.24	34	27	.26	26	28
>3.8 ≤ 4.1	24	.24	29	24	.27	23	25
>4.1 ≤ 4.3	22	.24	26	22	.27	21	23
>4.3 ≤ 4.5	20	.24	24	20	.26	19	22
>4.5 ≤ 4.7	18	.23	22	18	.26	18	20
>4.7 ≤ 5.0	16	.24	18	16	.26	15	18.6
	Volume Limits						
	L	fc	SIU	L	fc	SIU[1]	SIU[2]
>Nat. ≤ 3.5	31	.37	38	22	.37	25	25
>3.5 ≤ 4.1	25	.36	31	18	.35	21	21
>4.1 ≤ 5.0	22	.40	24	17	.39	17	17
	Cyl. Dia. Limits						
	in.	fc	SIU	in.	fc	SIU[1]	SIU[2]
>Nat. ≤ 3.5	10.7	.32	11.1	9.5	.33	9.5	9.5
>3.5 ≤ 4.1	9.8	.30	10.4	8.9	.31	8.9	8.9
>4.1 ≤ 5.0	9.2	.32	9.3	8.4	.31	8.4	8.4

---

(1) SIU limit based on optimum pellet size.

(2) SIU limit based on 0.30 inch diameter pellets

TABLE II-2.

Calculation of Methodology Standard Deviation.

Expt. No.	Av. Neut. Energy Grp	KENO K-eff	Keff Poly. Fit	Delta Keff	Diff.Sq'rd
1	14.8539	1.00292	0.99907	3.84575e-03	1.47898e-05
2	14.8634	0.99757	0.99896	-1.39096e-03	1.93476e-06
3	14.8664	1.00687	0.99893	7.94448e-03	6.31147e-05
4	14.8084	1.00672	0.99964	7.08029e-03	5.01305e-05
5	14.8107	1.00075	0.99961	1.13978e-03	1.29911e-06
6	14.8196	0.99834	0.99950	-1.15701e-03	1.33867e-06
7	14.7755	0.98474	1.00007	-1.53321e-02	2.35074e-04
8	14.7919	0.99528	0.99985	-4.57411e-03	2.09225e-05
9	14.9939	0.99028	0.99757	-7.29158e-03	5.31672e-05
10	15.1072	0.99605	0.99662	-5.67638e-04	3.22212e-07
11	14.4363	1.00986	1.00568	4.17660e-03	1.74440e-05
12	14.2973	1.00515	1.00859	-3.43996e-03	1.18334e-05
13	14.8391	1.00357	0.99925	4.31597e-03	1.86276e-05
14	14.7265	1.00633	1.00075	5.57716e-03	3.11047e-05
15	15.0905	1.00059	0.99674	3.84649e-03	1.47955e-05
16	15.1965	0.99362	0.99603	-2.41102e-03	5.81302e-06
17	15.3165	0.99081	0.99547	-4.66215e-03	2.17356e-05
18	15.2583	0.98069	0.99571	-1.50203e-02	2.25611e-04
19	15.5059	0.99957	0.99513	4.44468e-03	1.97552e-05
20	15.4858	0.99628	0.99513	1.14896e-03	1.32011e-06
21	14.3444	1.02009	1.00757	1.25245e-02	1.56862e-04
22	14.3683	1.01826	1.00706	1.11988e-02	1.25413e-04
23	14.3416	1.01304	1.00763	5.41469e-03	2.93189e-05
24	14.3310	1.00997	1.00785	2.11714e-03	4.48226e-06
25	14.3567	1.01011	1.00730	2.80531e-03	7.86978e-06
26	14.1106	1.00180	1.01305	-1.12494e-02	1.26549e-04
27	14.1011	1.01007	1.01329	-3.22333e-03	1.03899e-05
28	14.1039	1.00609	1.01322	-7.13126e-03	5.08549e-05
29	15.0206	1.00216	0.99733	4.83434e-03	2.33709e-05
30	14.8915	0.99504	0.99864	-3.59551e-03	1.29277e-05
31	14.8981	0.99901	0.99856	4.48843e-04	2.01460e-07
32	14.8881	0.99146	0.99867	-7.21412e-03	5.20435e-05
33	15.4248	0.98880	0.99519	-6.39360e-03	4.08781e-05
34	15.4362	0.99287	0.99518	-2.30674e-03	5.32107e-06
35	15.4263	0.98804	0.99519	-7.15125e-03	5.11403e-05
36	15.4365	0.98847	0.99518	-6.70633e-03	4.49749e-05
37	15.5075	1.00691	0.99513	1.17848e-02	1.38882e-04
38	15.5096	0.99887	0.99513	3.74493e-03	1.40245e-05
39	15.5233	0.99687	0.99513	1.74368e-03	3.04041e-06
40	15.5514	1.00548	0.99514	1.03404e-02	1.06923e-04
				Sum	1.81560e-03
				Variance	4.65539e-05
				Std.Dev.	6.82304e-03

Table II-3

KENO IV RESULTS FOR NOTED GAP WIDTHS

Description	Hydrogen Density gm/cm^3	KENO IV K_{eff}
	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

Figure II-6

KENO K_{eff} Including Uncertainty and Bias, Versus Kilograms Uranium Per Liter for Mass Limited Unit Containers

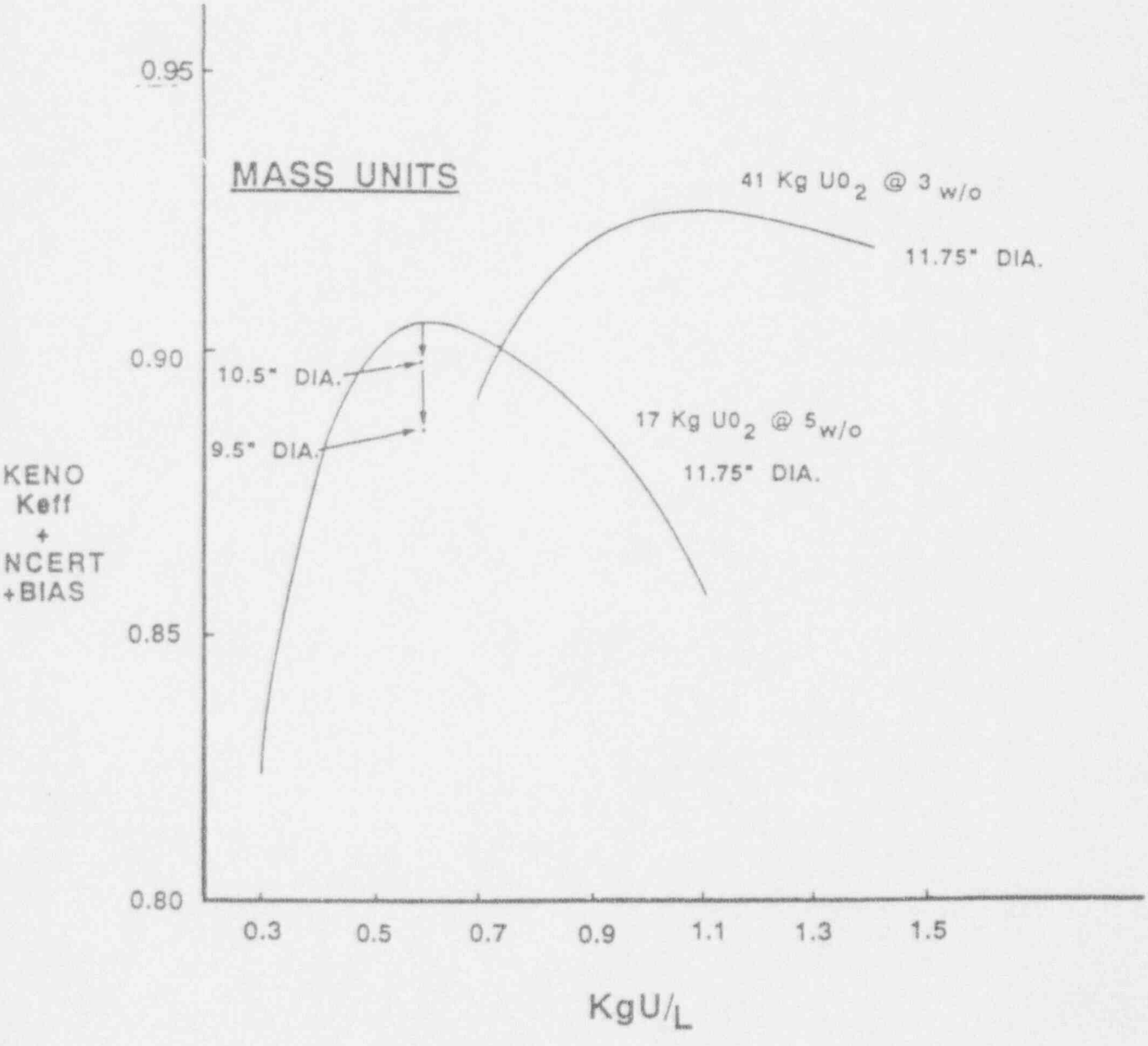


Figure II-7

KENO K_{eff} Including Uncertainty and Bias Versus $ts/_{tc}$ and Kg U/L for Cylinder Diameter and Volume Limited Unit Containers

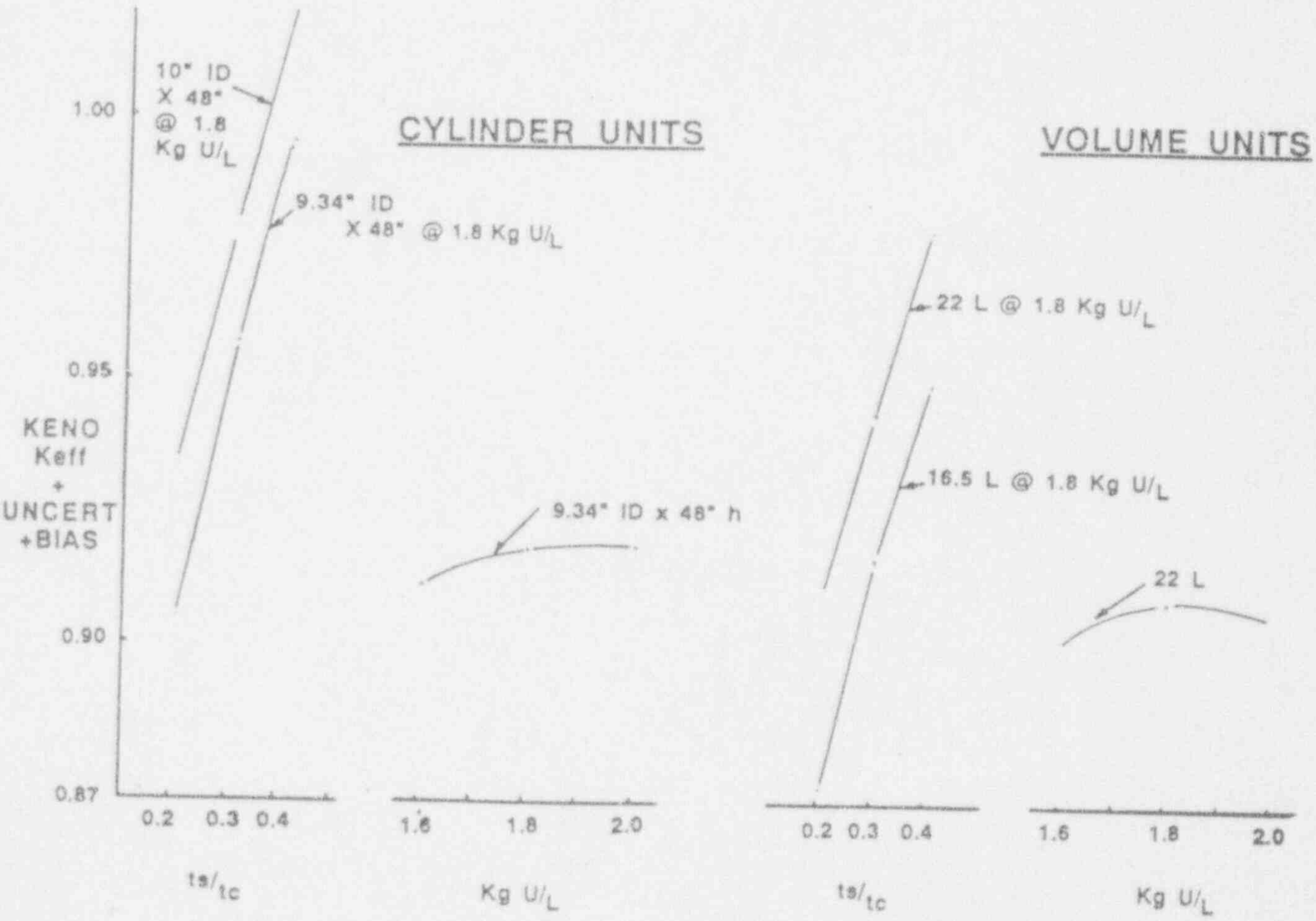


FIGURE II-8

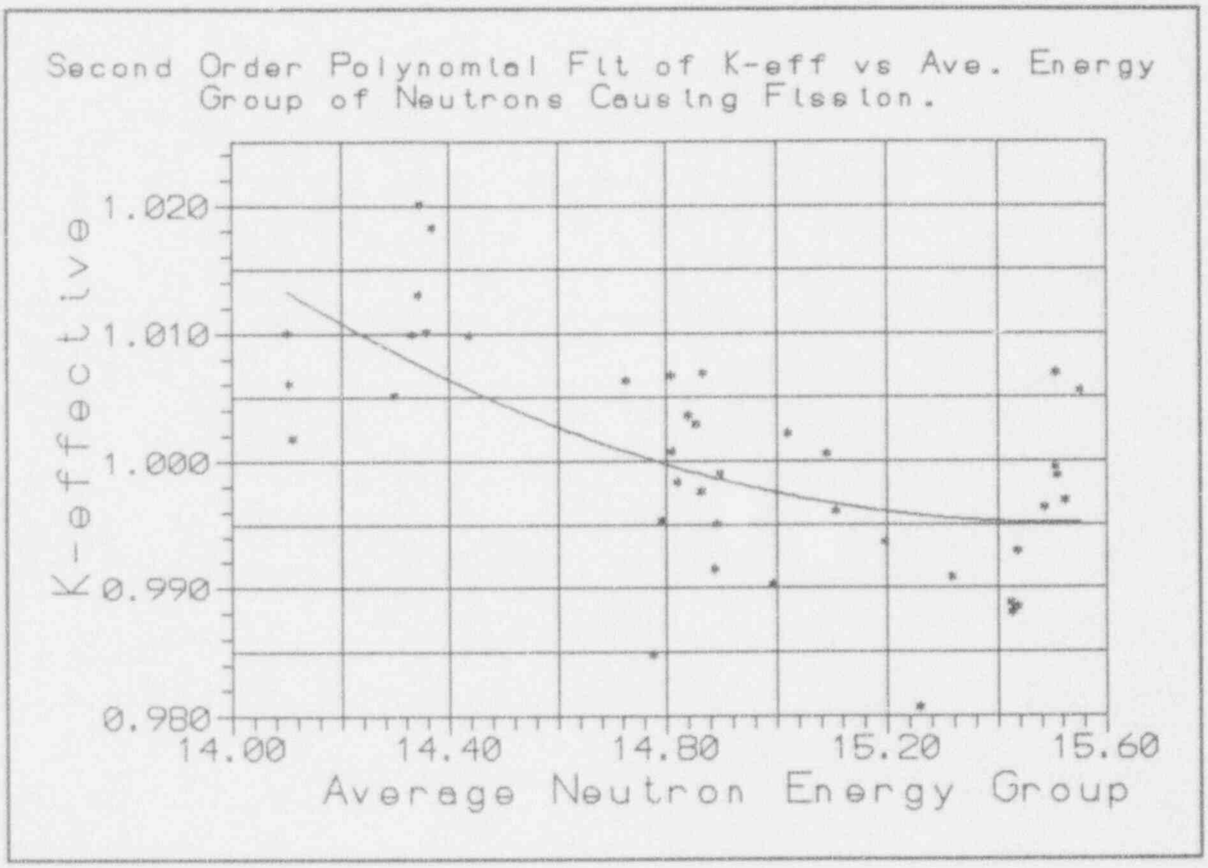
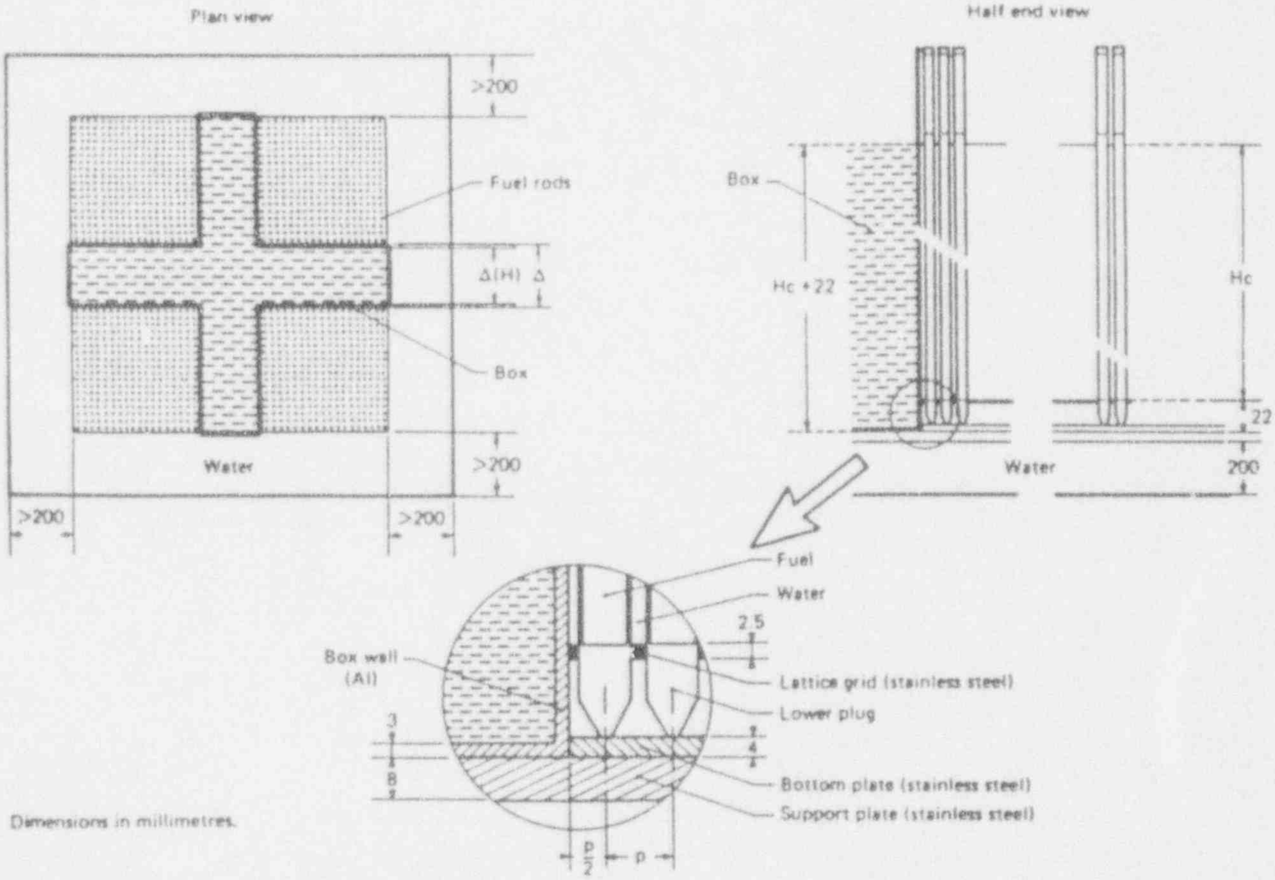

Second Order Polynomial Fit of K-eff vs Ave. Energy Group of Neutrons Causing Fission.
K-effective
1.020
1.010
1.000
0.990
0.980
14.00
14.40
14.80
15.20
15.60
Average Neutron Energy Group

Figure II-9

EXPERIMENTAL SETUP FOR EXPERIMENTS
OF REFERENCE 6



8.0 OPERATIONS DESCRIPTION AND SAFETY ANALYSIS

This section contains descriptions of SNM handling and storage operations during the Decontamination and Decommissioning of Buildings 17 and 21. Nuclear criticality safety limits are consistent with the limits of Chapter 4 of Part 1.

8.1 SNM STORAGE

SNM is stored in covered steel containers having a volume of 5 gallons, or less. These containers are stored in either: 1) a designated storage array employing the criteria of Section 4.2.2.1 of Part I, or 2) within UNC-2901 shipping containers (Certificate of Compliance No. 9264). The storage array of 5 gallon, or less, containers employs positive spacing devices to assure the spacing area is preserved.

8.2 PRE-TREATMENT OF LOW LEVEL LIQUID WASTES AND NUCLEAR SAFETY

8.2.1 Pre-treatment of Low Level Liquid Wastes

Aqueous wastes from low level radioactive cleanup operations such as mop buckets and decontamination solutions are processed through a system designed to remove particulate matter. Figure II-10 shows a sketch of the liquid waste processing system layout in the Building 17 Annex.

Contaminated liquids are batch processed by this system until sampling checks verify that the activity level is below a specific threshold prior to pumping to the Building 6 liquid holding tanks. Solids removed are handled and processed as residual uranium oxide material. The overall process is depicted in he flow chart of Figure II-11.

8.2.2 Nuclear Safety

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

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

(2) The dump tank has a capacity of 5 gallons or 18.9 liters. This value is 26% less than the critical, fully reflected volume of 25.5 liters inferred from the most conservative data of Figure II-12 for 0.050 inch diameter pellets. In the event that the prefilter failed, the dump tank is still 15% less than the critical, fully reflected volume of 22.1 liters for the optimum pellet diameter of 0.3 inch diameter pellets. The bottom of the dump tank is about nine inches off the floor; consequently, the likelihood of full reflection is small, as is the likelihood of having 0.3 inch diameter pellets uniformly distributed throughout the volume of the dump tank with a water to oxide volume ratio of about 2.8.

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

Under normal operating conditions, the concentration of UO_2 in the slant tank is sufficiently low that it is impossible to achieve criticality regardless of the tank volume or geometry. The concentration of UO_2 in the slant tank only approaches that of the solution being poured into the dump tank when the sludge volume of the centrifuge bowls approaches full capacity. Nevertheless, the slant tank geometry is set so as to preclude criticality in the event that the slant tank is fully reflected and filled with a uniform distribution of optimally sized UO_2 pellets at optimum moderation.

The most conservative data of Figure II-12 on critical slab thickness for an optimally moderated and reflected slab versus particle size shows that the minimum slab thickness occurs for particles/pellets having a diameter of 0.2 to 0.4 inches. The critical slab thickness is 4.15 inches. The corresponding water to oxide volume ratio is about 2.3. If the presence of the screen barrier is assumed to reduce the reflection of the tank by 50%, this is equivalent to an increase in the critical slab thickness of 1.6 inches for optimum moderation conditions within the tank. The slant tank is approximately four feet off the floor; consequently, flooding of the surrounding area so as to reflect the tank is highly improbable. An approach to criticality, even under the postulated failure of the prefilter and centrifuge to remove UO2 particles from the solution entering the slant tank, cannot occur as long as the slant tank thickness is maintained.

The slant tank engineering design is such that dimensional changes with postulated loading of the tank are minimized. In the absence of any structural supports other than at the edge of the tank and calculating the deflections for two coupled (via five internal braces) 1/8 inch thick plates, it was estimated that the deflection resulting from a mass distribution of liquid of density 1 g/cc (pure water) would be 0.128 inches; for a contaminated solution density of 2.5 g/cc, the deflection is estimated at 0.249 inches; and for a solution density of 3.5 g/cc, the deflection is 0.333 inches. To minimize deflections, the 1.5 x 1.5 x 0.25 angle braces were run diagonally along the lower face of the tank and a central support leg to the floor was added.

In summary, the slant tank is structurally reinforced and vented so as to minimize possible deflection of the tank and enlargement of the liquid slab thickness. The critical slab thickness for optimum moderation and particle size conditions within the slab tank, assuming half reflection of the tank, is conservatively estimated as 5.75 inches. This value is based on using the most adverse data of Figure II-12 as well as the critical buckling and reflector savings data of DP-1014 for 5 w/o enriched UO_2. This derived value exceeds the 4.0 inch maximum design thickness criterion for the slant tank by 44 percent.

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

(5) The centrifuge hood is a three part hood. The central section is occupied by the centrifuge. The right and left sections are designated as mass limited regions; the contaminated scrap is handled under the SIU mass limits defined in Chapter 4 of Part I. The central region of the hood containing the centrifuge has a floor that is below the right and left work surfaces by about 20 inches. However, this well type area is drained by a line going to the overflow tank. It is also noted that a city water line enters the hood but the valve is exterior to the

hood; thus, should the line break within the hood, it would not flood the hood.

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

8.3 TRANSFER OF MATERIAL

Material may be transferred on carts which accommodate one mass limited, one volume limited container, or one slab limited array, or transferred by hand. For mass or volume limited containers, the cart shall provide for centering of the container on the cart and the cart shall be of such dimensions that the required spacing area is provided. The transport carts shall not cross spacing areas provided for process operations unless material is being moved to an available storage position.

8.4 HOODS

Two types of hoods may be employed beyond that provided for the centrifuge.

1) General Purpose Hood- This is a ventilated hood which is used for miscellaneous work involving the handling of UO_2 powder or contaminated material.

2) Filter Knockdown Hood- This is a ventilated glove box hood which is used, for example, to remove loose UO_2 from used absolute filters and prefilters.

Individual hoods shall use mechanical devices to assure a one foot separation between containers if more than one container is permitted within the hood.

FIGURE II-10
CENTRIFUGE COMPLEX LAYOUT IN BUILDING 17
(DIMENSIONS ARE APPROXIMATE)

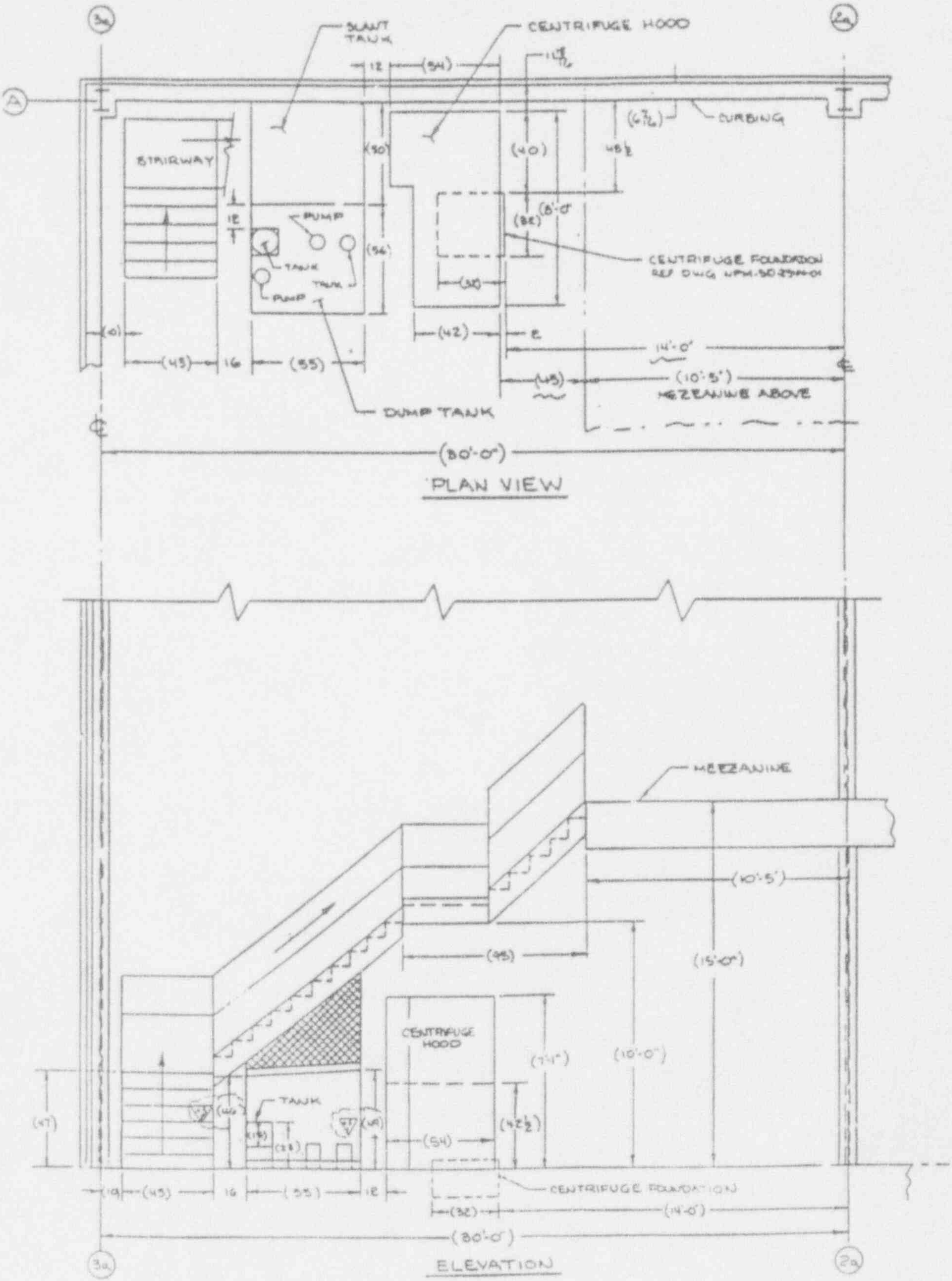


FIGURE II-11

FLOW CHART FOR BUILDING 17 CENTRIFUGE COMPLEX

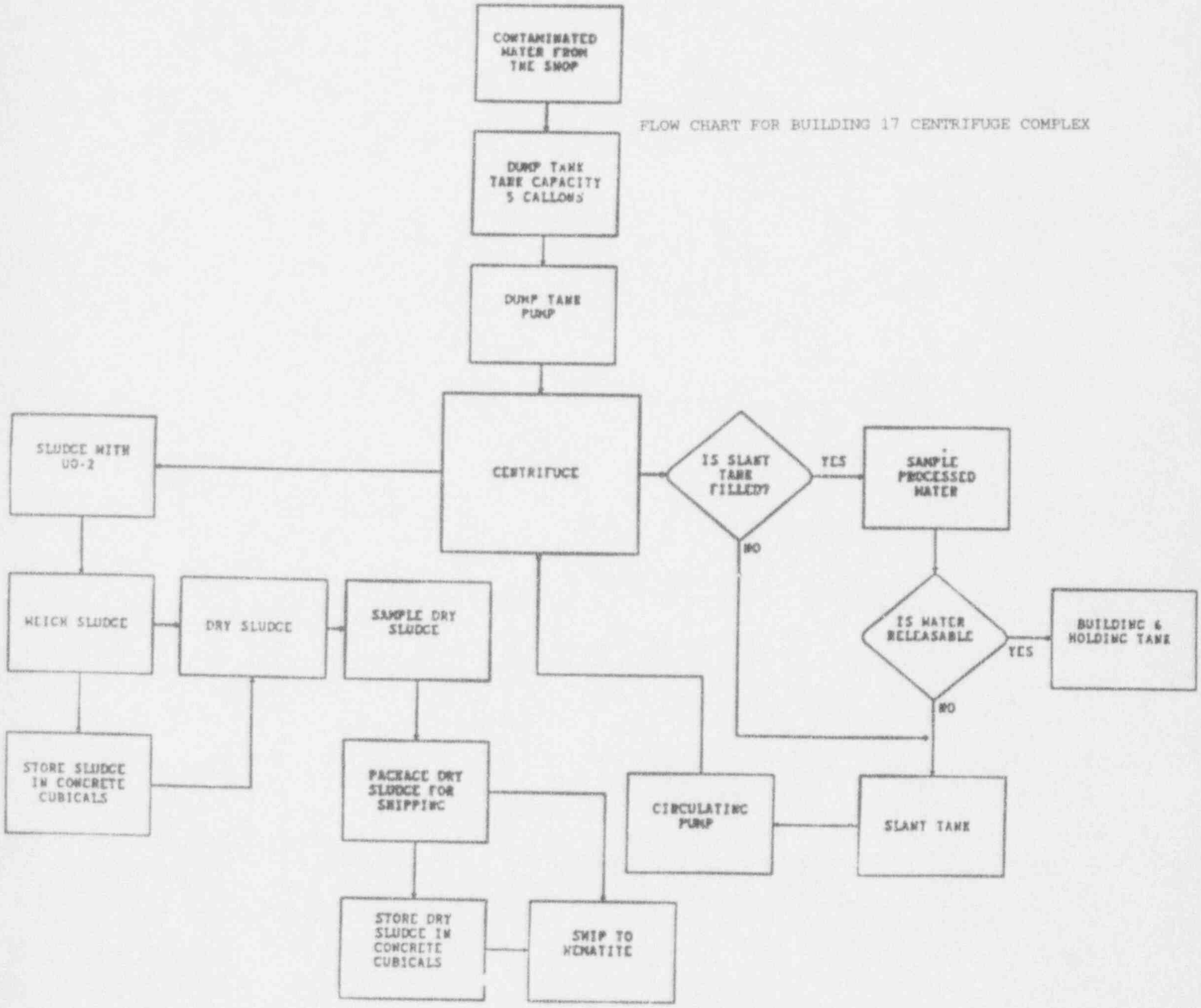
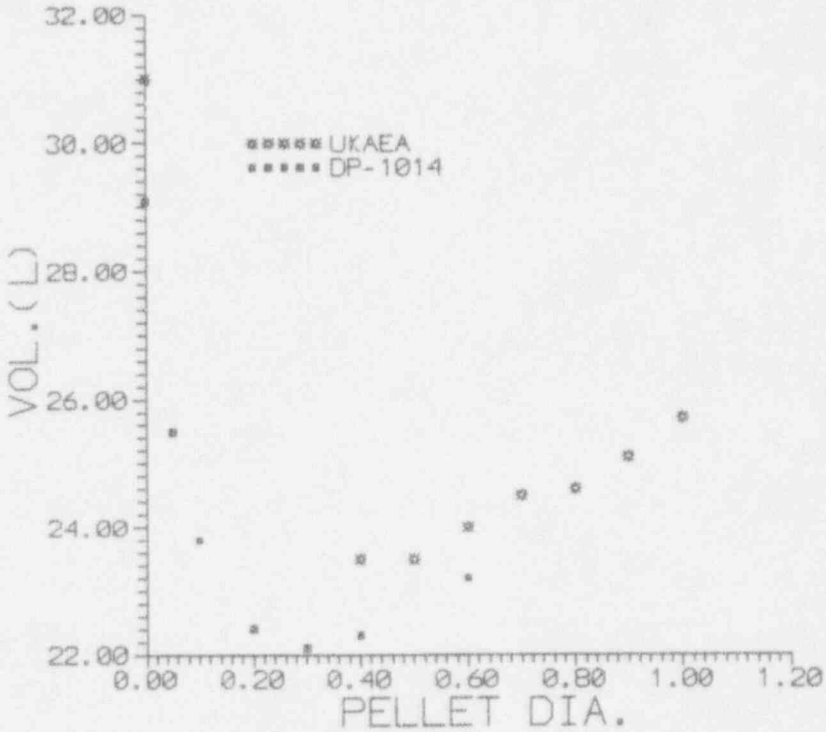
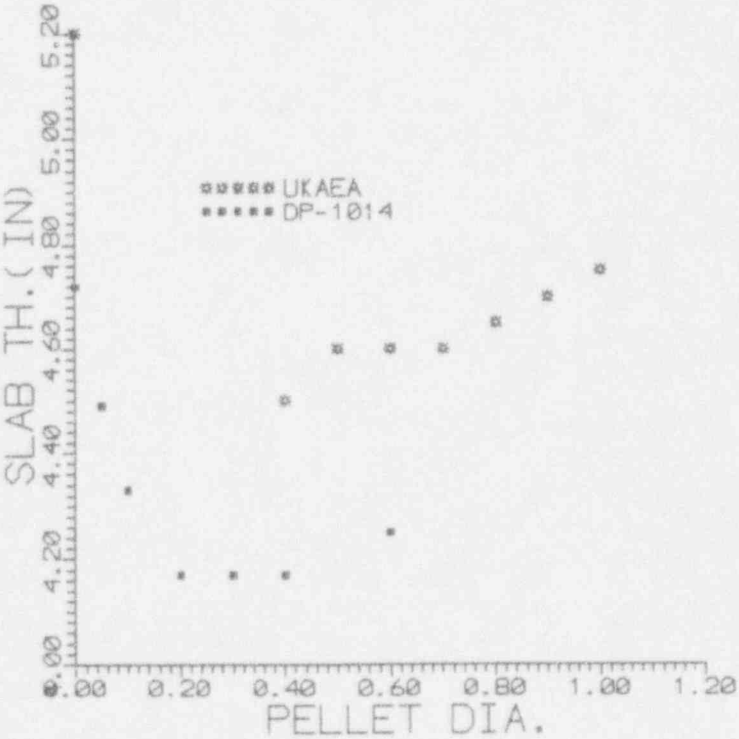


FIGURE II-12



32.00
30.00
28.00
26.00
24.00
22.00
VOL.(L)
UKAEA
DP-1014
0.00
0.20
0.40
0.60
0.80
1.00
1.20
PELLET DIA.



5.20
5.00
4.80
4.60
4.40
4.20
SLAB TH.(IN)
UKAEA
DP-1014
0.00
0.20
0.40
0.60
0.80
1.00
1.20
PELLET DIA.