R3/D2 #45

ABB

ASEA BROWN BOVERI

October 2, 1992
ML-92-045

Docket No. 70-36
License No. SNM-33

Mr. John W. Hickey, Chief
Fuel Cycle Safety Branch
Division of Industrial and Medical Nuclear Safety
Office of Nuclear Materials Safety and Safeguards
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Response to NRC Questions on the Hematite License Renewal Application

Reference: (A) Letter, M. Tokar (NRC) to J. A. Rode (C-E), dated July 10, 1992

(B) Letter, J. F. Conant (C-E) to J. W. Hickey (NRC), ML-92-041, dated August 31, 1992

Dear Mr. Hickey:

Reference (A) requested additional information regarding the Hematite License Renewal Application submitted November 1989 and supplemented June 1991. Reference (B) presented Combustion Engineering, Inc.'s two-part plan for submitting the requested information. The enclosures to this letter provide a substantial portion of the additional information you have requested and represents our first response submittal.

Enclosure I to this letter provides the NRC questions followed by our response. Responses which will be provided in the second submittal package have been indicated as such. Enclosure II provides a listing of the License Renewal Application change pages. Enclosure III provides the actual renewal application change pages for insertion into your copies. Six (6) copies of this document are provided for your use.

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 6
FOIA- 2004-0234

J J-H

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The enclosed responses represent the majority of the requests for additional technical information. The balance of the requested information should be provided in approximately one month. If there are any questions or comments concerning this matter, please do not hesitate to call me or Mr. Mark A. Michelsen of my staff at (203) 285-5261.

Very truly yours,

COMBUSTION ENGINEERING, INC.

John F. Conant

John F. Conant
Manager
Nuclear Materials Licensing

JFC:cr

Enclosures: As Stated

cc: G. France (NRC - Region III)
S. Soong (NRC)

Enclosure I to
ML-92-045

COMBUSTION ENGINEERING, INC.

HEMATITE NUCLEAR FUEL MANUFACTURING FACILITY

RESPONSE TO NRC QUESTIONS ON THE

LICENSE RENEWAL APPLICATION

INDIVIDUAL RESPONSES

October 1992

Request for Additional Information
Application Dated November 22, 1989
Combustion Engineering, Inc.
Docket No. 70-36

General

1. On March 21, 1989, the NRC published the "Guidance On Management Controls/Quality Assurance, Requirements for Operation, Chemical Safety, and Fire Protection for Fuel Cycle Facilities" 51 Federal Register 11590 (attachment 1). CE should evaluate its safety program in accordance with this guidance, propose license conditions, and commit to addressing these conditions within 1 year. If additional time is required, provide justification.

   Response:

   The response to this question will be provided in the second submittal.

2. Modify the license conditions section to incorporate all appropriate license amendments that were issued. Include the license conditions that were imposed in the amendments and commitments made in support of the amendment application.

   Response:

   The License Renewal Application has been revised to incorporate all appropriate license amendments that have been approved (through Amendment #20). Change pages are enclosed. In addition, the following Conditions from the Materials License have been incorporated into the Renewal Application.

   CONDITIONS INCORPORATED INTO THE LICENSE APPLICATION

Condition No.	*Section No.*
12	*3.1.1*
14	*3.2.6.2*
15	*1.6(a)*
20a	*5.1.2*
21	*Chapter 8*

CONDITIONS INCORPORATED INTO THE LICENSE APPLICATION

Condition No.	*Section No.*
22	*Chapter 7*
31	*4.2.3.2(a)*
32	*4.2.1.1;4.2.2*
33	*2.6(b)*
39	*4.2.4(i)*

The following additional Conditions are scheduled to be addressed in the second submittal: No. 13, 16, 17, 18, 19, 20b, 23, 24, 25, 26, 27, 28, 29, 30 and 34 (Conditions 11, 35, 36, 37 and 38 have previously been deleted).

3. NRC staff has determined that the licensee should establish greater formality in programs related to nuclear criticality safety (NCS). The application should provide further details of management programs and the administrative and operational requirements stemming from these programs, such as safety analyses, configuration control, maintenance and surveillance, training, and audits. The programs should be documented by written policies, procedures, or instructions. The programs should provide control over activities affecting the safety systems.

   Many of these requests require only formalization and documentation of existing practices into auditable programs. Commitment to programs and administrative and operational requirements should be in Part I of the application. Discussion and description of programs may be summarized in Part II of the application provided internal documentation can be referenced for detailed information.

   Some of the requests for additional information will require significant commitment to time and resources. Completion of all requests is not required for renewal. Some replys may propose license conditions accompanied with conditional phrases specifying completion within a particular timeframe. Such license conditions may be required for issues related to the safety analysis, configuration control, and maintenance and surveillance. Separate discussions for these topics are enclosed with this request (attachment 2).

   Response:

   The response to this question will be provided in the second submittal.

Specific

Page Comment

1-1

A. Section 1.3 may be revised to request a 10-year license as NRC has allowed.

Response:

Section 1.3 has been revised to indicate a license renewal period of ten (10) years.

B. The section should be revised to add that at not more than 1-year intervals from the license issuance date, the demonstration section will be updated to reflect the current operations as appropriate. The updates should, as a minimum, include information for the health and safety section as required by 10 CFR 70.22(a) through (f) and 70.22(i), and operational data, and information on environmental releases.

Response:

C-E has added in Section 1.6 of Part I of the renewal application a commitment to the affect that updates will be provided at 2 year intervals from the date of renewal approval, except that the renewal application at the end of the 10 year renewal period may replace that update.

It is C-E's position that a two year update frequency is satisfactory when considering the types of low level activities carried out at the facility and the types of changes that are anticipated to occur over the next ten years. Furthermore, a two year update frequency is consistent with the staff's proposed rulemaking (57 FR 27187, June 18, 1992) which would allow a period of up to two years for Final Safety Analysis Report (FSAR) Updates.

<u>Page</u> <u>Comment</u>

1-2 Section 1.5 should describe the activities that use uranium enriched greater than 5 weight percent in U-235 and the licensed activities in Building 110 and 240-1. Identify the location(s) where the Co-60 and mixed activated and fission product calibration sources will be stored and used. In describing the utilization of each building, reference Figure 9-4.

Response:

Building 110 and 240-1 are not normally places where large quantities of SNM are used. At times however, samples may be present in this area for miscellaneous purposes. Calibration sources or laboratory standards may be used anywhere within the facility provided they are handled following proper procedures, and by authorized personnel. We do not think it is appropriate to reference a Figure from Part II of the license application in this section.

1-3 Section 1.6 discusses disposal of radioactive waste by incineration. In Part II, provide the information in attachment 3, "Information Required for Approval of Disposal by Incineration."

Response:

Following are responses to the questions presented in Attachment 3, "Additional Information Required for Approval of Disposal by Incineration".

1. *Section 15.7.1.2, Waste Incineration, has been revised to include the information requested.*

2. *The limits specified in Appendix B, Table II of 10 CFR Part 20 are for assessment and control of dose to the public and are usually applied at the site boundary. Combustion Engineering controls these limits at the stack, which demonstrates compliance with the ALARA philosophy.*

3. *The concentration of radioactive material (uranium) in the ash is determined by withdrawing a representative sample from each container of ash and analyzing for percent uranium.*

4. *Procedures are described in Sections 10.4.2 and 15.7.1.2. Packaging of combustibles in plastic bags, use of ventilated hoods to prepare charges for incineration, and ash removal by a vacuum collection system limit exposure of personnel.*

<u>Page</u> <u>Comment</u>

5. *Combustion Engineering currently complies with state and local regulations concerning incineration of radioactive material. The state of Missouri has promulgated new regulations on incinerators which are scheduled for implementation during 1993. Modifications to increase the temperature of the secondary combustion chamber are anticipated in order to comply with these regulations when they become effective.*

6. *Fire safety controls are discussed in Sections 10.6 and 15.7.1.2.*

2-1 Chapter 2 should be revised to include the organizational changes authorized by Amendment 20.

Response:

Chapter 2 has been updated to incorporate the changes authorized by amendment #20. Change pages have been provided.

2-3

A. In Section 2.3, formal review and approval for process, equipment, and procedural changes should involve the Safety Review Committee. Conditions for which NRC approval is required should be established.

The licensee may make facility, structural, process, equipment, and procedural changes without license amendment provided that any proposed change does not involve (i) a modification to the conditions of this license or Part I of the referenced application; (ii) a significant increase in occupational radiation exposures; (iii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (iv) an unreviewed safety question. For facility, structural, process, equipment, and procedural changes not requiring a license amendment in accordance with the above criteria, an evaluation should be required. Such evaluations should be reviewed and approved by the safety manager and Hematite Plant Safety Committee.

The evaluations should provide the basis for determining that the change will not involve a modification to the conditions of this license or Part I of the referenced application, a significant increase in occupational radiation exposures, a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or an unreviewed safety question. A change should be deemed to involve an unreviewed safety question if an accident analysis for the change (i) results in

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consequence values exceeding the values of the accident analyses described in Chapter 16 of the referenced application or the probability of occurrence for the types of events therein evaluated is judged to increase; or (ii) reveals a possibility for an accident of a different type than previously evaluated. The licensee should maintain records of approvals and evaluations of facility, structural, process, and equipment changes until termination of the license. Records of procedural changes should be maintained for a minimum of 5 years.

Response:

The response to this question will be provided in the second submittal.

B. Within Section 2.5, requirements should be established for the training program and should include: (1) responsibilities for development, implementation, and coordination of NCS training; (2) NCS staff participation in development and implementation of the training program; (3) retraining following revision to equipment, processes, or operating procedures (retraining should be conducted prior to operation of installed equipment or use of revised procedures); (4) training for supervisors, maintenance personnel, engineers, NCS staff, management, and the Safety Review Committee; (5) assessing training effectiveness; (6) auditing the training programs at least annually; (7) updating training courses to reflect plant modifications and changes to procedures; and (8) troubleshooting activities for process abnormalities in operations training.

Response:

Regulatory Guide 3.52 describes the content requirements for Section 2.5 of the License Renewal Application. Our current discussion on "Training" in the renewal application complies with the suggested content of this Regulatory Guide. In Section 2.5, C-E has made a commitment to provide training to employees on radiological safety, criticality safety, and special skills. Employees are retrained biennially in radiological and criticality safety, and tested. This training is documented.

We have added training responsibilities in Section 2.1.2 for the Manager, NLS&A position as emphasis to the importance of safety training.

With regard to the Nuclear Criticality Specialist, Section 2.2 and Table 2-1 currently require that the person filling this position have a Bachelor's Degree in Science or Engineering, plus at least 2 years experience in nuclear

<u>Page</u> <u>Comment</u>

criticality evaluations. The education and experience requirements should ensure sufficient training for the NCS.

2-4 In Section 2.6, Operating Procedures, the excerpt "limits and controls required by the license" should read "limits and controls identified in the NCS evaluations."

Requirements should be established for the contents of operating procedures. The contents should include process operating limits, sequence of steps to be taken under upset conditions, safety systems and functions, precautions, and warnings. The procedures should address all aspects of operation including startup, temporary operation, and shutdown. Instructions and criteria for shutdown and actions to be taken during abnormal operations should be specified, including the limits selected for a commitment to action.

Further requirements should be established for developing, approving, and updating procedures. Supervision should be involved with the development of operating procedures. Biennial review of procedures should include line and nuclear safety management. The procedures should be approved at least by the NCS Function Manager and the Operating Group Manager. The approval process should be established by the plant manager. New or revised procedures should be reviewed by NCS staff. When process and equipment changes occur, changes to procedures should be preceded by a safety analysis, management approval, preoperational testing and inspection, and training.

Response:

The response to this question will be provided in the second submittal.

2-5 Management should examine the manageability of programs related to NCS, establish management controls to monitor the programs' effectiveness, and ensure adequate implementation. The status and adequacy of the programs should be reviewed at least biennially. Thus, within Section 2.7, a requirement should be established for assessing management programs and policies. A safety oversight group should assess the manageability, implementation, and effectiveness of programs instituted for audits and inspections, corrective actions, design basis documentation, maintenance and surveillance, training, configuration control, and safety analyses. The assessment should include an intensive and systematic examination and

Page Comment

should be conducted by a team with multidisciplined personnel possessing the expertise necessary for proper review of the programs.

An action plan, including follow-up and tracking, should be developed to address concerns resulting from the self-assessment. Items requiring corrective action should be documented in a report to management. The program should possess requirements to follow-up the report. The follow-up should determine completion of corrective action and document resolutions to deficiencies. The follow-up actions taken by the responsible manager should be documented.

Requirements established for audits and inspections should include: guidance provided for conducting audits; the format (procedure or checklist), staffing, scheduling, and verification methods prior to conducting the audits; responsibilities for root cause analysis, designating corrective actions, tracking, and documentation; the system to ensure corrective action; and the level of management to which results are reported. Corrective actions and their status should be maintained in the audit records.

Requirements should be established to ensure that the periodic review of existing processes includes verification of the conditions and assumptions used in the safety analyses and absence of unapproved alterations to processes, equipment, or procedures. Efforts in review of engineered controls should involve evaluation of the programs established for maintenance, surveillance, and functional testing. Requirements should be established for auditing the maintenance and surveillance of engineered controls.

Response:

The response to this question will be provided in the second submittal.

2-6 Within Section 2.8, the investigation program should be further established and include provisions for root cause analysis and tracking of corrective actions.

Requirements should be established for incident investigation procedures that address such issues as team members, reporting, information dissemination, recommendations, and incident pattern; training team members in investigation techniques; tracking and correcting identified deficiencies; and a system to promptly address and resolve the incident report findings and recommendations. Problem identification, reporting,

Page Comment

resolution, tracking, trending and root cause analysis systems should be adequately developed to allow management to monitor corrective actions. There should be a formal management program to evaluate operating experiences and improve safety.

Response:

Section 2.8, "Investigations and Reporting" currently states that reportable events shall be investigated and reported and non-reportable events shall be investigated and documented as appropriate. The level of investigations performed in order to understand the cause of the incident is proportionate to the severity or potential severity. Not every incident will require proceduralized root cause analysis.

The intent of investigations and reporting is the same as the intent of the Branch Technical Position on Management Controls, i.e., to identify items important to safety, recognize their potential significant failures and provide feedback to assess management program effectiveness. It is more appropriate for the prescriptive actions described above to be implemented on a case by case basis than to make them conditions of the license.

3-1

A. In Section 3.1.2, review the Special Evaluation Traveler for industry safety. A member of the radiation safety and protection functions shall monitor the work areas under a Special Evaluation Traveler.

Response:

Section 3.1.2 states that the same approvals are required for Special Evaluation Travelers as for Operation Sheets. Section 2.6 states that the Manager of NLS&A (among others) approves Operation Sheets, and Section 2.1.2 states that the Manager, NLS&A manages industrial safety. Therefore, a further commitment to review Special Evaluation Travelers for industrial safety should not be required.

The Special Evaluation Traveler is used for all operations not covered by an Operation Sheet. Radiological monitoring is not always required for the operation; more often than not, the Special Evaluation Traveler does not require special monitoring. If special monitoring is required, the Special Evaluation Traveler specifies the requirements. Therefore, Section 3.1.2 has not been revised to specify that work areas under a Special Evaluation Traveler be monitored by Health Physics.

Page Comment

B. In Section 3.2.1, indicate that a routine review will be conducted to verify that signs, labels, and other access controls are properly posted and operative. The review should be documented. A minimum frequency for the review may be specified in Part II of the application.

Response:

This requirement is satisfied through the inspections/audits conducted in accordance with Section 2.7 of the license application. Additionally, Health Physics technicians review access controls during their daily rounds of the facility.

3-2

A. Section 3.2.2 should state the following:

(1) In process areas, the HEPA system shall be equipped with an indicator for pressure-drop across the filter(s) to provide an early indication of a reduction in air flow. The pressure reading should be checked at least weekly. Deviation from this requirement should be justified.

Response:

Weekly air velocity measurements are made for ventilated hoods to ensure adequate air flow. This procedure checks the entire ventilation system, including the HEPA filters. HEPA filter and pre-filter banks are provided with differential pressure gauges for diagnostic purposes. Section 3.2.2 has been revised to describe this.

(2) The HEPA filter shall be replaced when the pressure differential across the filter exceeds 4 inches of water or the manufacturer's recommended level.

Response:

Filters or pre-filters are normally changed if the differential across the filter exceeds six inches of water. Experience with multi-bank, multi-filter systems equipped with pre-filters has demonstrated that this value is appropriate. Section 3.2.2 has been revised to specify this differential pressure limit.

(3) The ventilation system shall be in-place leak tested after each HEPA filter replacement or after completion of major repair work.

Page Comment

Response:

Ventilation systems in the Oxide Conversion Building and New Pellet Plant (Building 254) are DOP (Dioctylphthalate) tested in place after any disturbance of the HEPA filters. New systems with this provision for testing will be installed by the end of 1993 in Building 255 and Building 240. Section 3.2.2 has been revised to specify that ventilation systems capable of being DOP tested will be tested after any disturbance, and that new HEPA systems will have DOP testing provisions.

(4) The direction of air flow in the process buildings shall be checked at least monthly and documented. If the air flow direction is not acceptable, action shall be taken.

Response:

The direction of air flow will be checked on annual basis in accordance with the new regulatory guide on air sampling. Section 3.2.2 has been revised accordingly.

(5) The specific glovebox pressure differential between the glovebox and the work area and the frequency for checking it.

Response:

The Hematite facility is eliminating use of static pressure glove boxes, therefore it is not necessary to add this requirement to the license application. Section 3.2.2 has been revised to delete the discussion on glove boxes.

B. Section 3.2.2 should clarify if the air in the processing areas will be recycled. If so, a monitoring program should be established to control the spreading of contaminated air.

Response:

The response to this question will be provided in the second submittal.

3-3

A. In Section 3.2.3.1, an alarm setpoint should be established for the continuous air monitors to provide an early warning of unexpected releases in the work areas. You should state that a means for measuring the flow

Page Comment

rate (such as rotameter or critical orifice) will be installed at each fixed air sampling head.

Response:

The response to this comment will be provided in the second submittal.

B. In Section 3.2.3.2, item (b) should be revised to state that when the individual's internal exposure (MPC-hours) exceeds 20 percent of 10 CFR 20.103 limits, corrective actions to the cause and personnel exposure evaluation will be required. Item (c) should be deleted because it is repetitive. Item (d) should state that an evaluation of the individual's internal exposure to airborne radioactivity should be based on breathing zone sampling data, which is obtained by continuous sampling during his/her presence in a work area where unclad radioactive material is handled. The survey frequency shall be in accordance with Table 1 of Regulatory Guide 8.24. Item (f) should state that the locations of the fixed air sampling heads shall be reexamined for representativeness at least every 13 months or whenever licensed process or equipment changes are made or at the commencement of operations in an area that has been shut down for more than 6 months, whichever comes first. Item (g) should state that during operations, the airborne radioactivity concentration in the process areas shall be assessed by continuous air monitors to identify any unexpected concentration level of radioactive material.

Response:

The 10 CFR Part 20 criteria will achieve the exposure limits of this comment. While Section 3.2.3.2, item (b), has not yet been revised, it will be revised before the new 10 CFR Part 20 becomes effective.

Item (c) in Section 3.2.3.2 has been deleted in the enclosed page changes.

With respect to items (d), (f) and (g), responses will be provided in the second submittal.

3-4 Section 3.2.4 should state that the air flow or volume metering devices for the air sampling program shall be calibrated at least once every 6 months, with exception of permanently installed effluent monitors which may be calibrated once every 18 months. Provide the minimum detectibility for all measurement instruments, and state that the accuracy of the calibration sources should be as a minimum ± 5 percent of the stated value and

Page Comment

traceable to the National Institute of Standards and Technology (formerly the National Bureau of Standards).

Response:

The response to this question will be provided in the second submittal.

3-5

A. Section 3.2.5 should state that the personnel radiation exposure levels shall be reviewed at least monthly by the Health Physics function.

Response:

Personnel radiation exposure levels are reviewed periodically by the Health Physics Supervisor and the Manager, NLS&A. In times of greater plant activity or special evolutions, this frequency is more than in times of inactivity. The adequacy of the frequency is evidenced by the ability to maintain exposures ALARA. The frequency of this review should not be a specific requirement of the license since Section 3.2.5 already requires an investigation for exposures in excess of 25% of the applicable limit, and Section 3.1.1 requires a radiation exposure report every six months.

B. In Section 3.2.6, provide a date for decontaminating the areas adjacent to Building 240, 253, and 256.

Response:

The response to this question will be provided in the second submittal.

3-6

A. In Section 3.2.6.2, references to contamination limits for release of equipment and material should be deleted. This information is in Section 1.6.

Response:

The references to contamination limits has been removed from Section 3.2.6.2.

B. In accordance with columns 2 and 4, Table I, of Regulatory Guide 8.24, establish the frequency for surface contamination surveys. State that cleanup action shall be started no later than the beginning of the next

Page Comment

workshift when surface contamination exceeds the limits in Table 2 of Regulatory Guide 8.24.

Response:

The response to this question will be provided in the second submittal.

C. Section 3.2.6 should state that change areas, where personnel exit from the contaminated areas, will be surveyed daily for removable alpha contamination.

Response:

The frequency of surveys is inappropriate as a condition in Part I of the license renewal application. As currently stated in Section 3.2.6.2: "The frequency of survey depends upon the contamination levels common to the area, the extent to which the area is occupied, and the probability of personnel exposure." In lieu of a specific survey frequency condition for the change areas, we have revised Section 12.14 to describe the general practice of surveying the change area on a daily basis.

D. Since Regulatory Guide 8.11 does not apply to bioassay for highly soluble uranium material (i.e., UF_6 or UO_2F_2), in Section 3.2.7, establish a program for detecting the workers' intake of the highly soluble uranium compounds.

The program should include the following:

(1) Criteria for determining who is required to participate in the program.

(2) Frequency for bioassay, action levels, and action to be taken at each level.

(3) Criteria for determining when a diagnostic bioassay measurement should be initiated.

Response:

The response to this question will be provided in the second submittal.

E. Justification should be provided for an annual frequency of in-vivo lung counts rather than semiannual.

<u>Page</u> <u>Comment</u>

Response:

The response to this question will be provided in the second submittal.

F. Section 3.2.8 should clarify whether protection factors for respirator equipment will be used in estimating exposures to individuals.

Response:

Additional detail has been added to Section 3.2.8 regarding protection factors.

4-1

A. In Section 4.1.1, because of the conditional phrase, incorporation of the double contingency principle is not explicitly expressed. An unconditional process design philosophy should be established. Also, favorable geometry should be established as the preferred method of control.

Response:

Section 4.1.1, "Process Design Philosophy", has been updated to more explicitly address use of the double contingency principle and indicate favorable geometry is the preferred method of control.

B. In Section 4.1.2, no position in the license application has been assigned the responsibility for establishing policies and practices implementing the NCS requirements. Management personnel responsible for formulating and implementing NCS policy should be indicated.

A specific procedure has not been established that ensures management approval of designs in which favorable geometry is not used as the method for criticality control. Use of nuclear criticality controls, other than favorable geometry, should require documented justification and management approval.

Response:

Section 2.1.1 currently identifies the Plant Manager as the management position with overall responsibility for safe operation, including criticality safety. The responsibility of the Plant Manager therefore includes that of NCS policies and practices.

Page Comment

Sections 4.1.1b) and 4.1.2 have been revised to address the second part of this NRC comment.

C. In Section 4.1.3, requirements should be established to ensure appropriate documentation of each analysis and review and to identify the personnel responsible for documentation.

Requirements should be established to perform a formal and comprehensive multidisciplinary safety analysis. (See Safety Analysis Discussion in Attachment 2.)

Response:

The response to this question will be provided in the second submittal.

4-2

A. In Section 4.1.4, procedures and guidelines should be established for routine activities of the Nuclear Criticality Safety Function, including participation in inspections, audits, NCS evaluations, and NCS training programs. The routine activities of the NCS function should be performed in accordance with written procedures which have been approved by the NCS function manager.

The development, review, change, approval, and implementation practices for all facility operating, maintenance, and testing procedures should be established. Documentation that provides requirements and guidance for identification, format, review and approval, distribution, and control of procedures should be identified.

The Nuclear Manufacturing Program documentation system that describes administrative and technical procedures relating to nuclear criticality safety should be a commitment within Part I. Specific authorities, responsibilities, and duties should be defined in the written administrative procedures. Such procedures should prescribe methods for formulating, implementing, and changing management safety programs.

Response:

The response to this question will be provided in the second submittal.

Page Comment

B. In Section 4.1.7, additional requirements should be established for preoperational testing of new equipment or processes. Preoperational testing and inspection should be documented and maintained as a record. Documentation should include deficiencies identified in the engineered safety systems or tests, resolutions to the deficiencies, and any retesting performed. "Substantially modified process" should be defined. The NCS analyst or reviewer should participate in the preoperational inspection. Thus, within this section of the application, "NLS & A" should be "NCS analyst or reviewer" and the "and/or" should be "and".

Response:

Section 4.1.7, "Preoperational Testing and Inspection", has been revised to more clearly address the requirements for preoperational testing and inspection for new or modified processes.

4-3

A. In Section 4.1.8, internal procedures that will be used for evaluating NCS of new processes or changes to existing processes should be identified. "Appropriate" safety review should be defined. The program for ensuring preparation of safety analyses and documentation of facility design should be identified.

Response:

Section 4.1.8, "Criticality Safety Design", has been updated to address internal procedures used for evaluating the nuclear criticality safety of new processes or changes to existing designs.

B. Proposed conditions regarding the use of approved written procedures for activities related to NCS design should include configuration control. Requirements should be established to develop and implement program and procedures for configuration control. (See configuration control discussion in Attachment 2.)

Criteria for approving NCS controls should be specified.

Response:

The response to this question will be provided in the second submittal.

Page Comment

4-6

A. In Section 4.2.3, the calculational methods that have been validated should be identified, and the contents of the validation report should be summarized.

Requirements should be established to ensure all calculational methods used to provide safety limits have a method validation study, including range of applicability and biases.

Written criteria and procedures for developing and approving criticality data sources and validation techniques for criticality calculations should be identified.

Requirements should be established to institute a validation program to update computer codes by reconfirming mathematical operations following changes in the computer program.

Response:

The response to this question will be provided in the second submittal.

B. In Section 4.3.4, special controls should be specified for solution transfer from favorable to nonfavorable geometry vessels, preventing the accumulation of fissile material in process equipment, verifying the isotopic content of incoming cylinders, and backflow prevention.

Response:

Additional detail has been added to Section 4.2.4 to discuss special controls.

C. Requirements should be established for measurement control. Measurement techniques employed should be identified and the technical basis for their validity provided.

Response:

The response to this question will be provided in the second submittal.

Page	Comment
5-1	In Section 5.1.2, "150 Ci" should be 150 microcuries.

Response:

Section 5.1.2 has been revised to read "150 μCi".

6-1
A. In Section 6.2, propose license conditions to establish, document, and implement a fire protection program. The program should include:

(1) Maintenance of the fire protection equipment, including the fire water system, automatic alarm system, and portable extinguishers. Such maintenance shall be performed in accordance with the applicable industry codes (e.g., the NFPA codes).

(2) Quarterly fire safety reviews by the Plant Safety Review Committee and follow-up actions on the findings.

(3) Weekly fire safety audits by the Fire Safety Supervisor and follow up actions on the findings.

(4) Performance of an initial (within 6 months and thereafter) biennial fire hazard analysis of the facility. This should be performed by qualified fire protection professionals. Address the findings of the analysis and implement corrective measures, where necessary, within a reasonable time. Any major modifications of the facility or the processes should necessitate a fire hazard analysis.

(5) In addition to the Emergency Plan, within 6 months, the establishment and maintenance of a current Pre-Fire Plan.

(6) A fire brigade training program, including curricula, examinations, and records. Include provisions for annual refresher training.

(7) Maintenance of documentation to evidence performance of the above activities.

Response:

The response to this question will be provided in the second submittal.

Page Comment

B. Describe in Part II, the existing fire protection equipment and those planned for within 1 year. Approximate completion dates for the installation of new equipment should be given.

Response:

Section 10.5 has been updated to include a description of fire protection equipment that has been recently added to the facility. There are no plans to add further equipment to the existing Hematite facility within the next year. Fire protection equipment for buildings being added as part of the Consolidation Project is described in the Consolidation License Amendment.

C. In Section 6.3, the emergency electric generators should be tested for operability at least weekly.

Response:

There is no significant safety benefit of requiring frequent tests of the emergency generator for a low enrichment fuel manufacturing facility such as Hematite. Nevertheless, Section 10.2.1 describes the normal plant practice of weekly startup testing of the emergency generator. During weeks when the plant is shut down for holidays or extended maintenance, testing of the generators is not performed. We do not recommend that this test be a commitment in Part I of the license.

7-1 Update Chapter 7 and provide a decommissioning funding plan in accordance with 10 CFR 70.25. This regulatory requirement was addressed in our letters dated June 27 and October 17, 1991.

Response:

As indicated in M. Tokar (NRC) letter to J. A. Rode (CE), dated February 26, 1992, the scheduled submittal on or about December 31, 1992 of a decommissioning funding plan for the Hematite Fuel Facility is acceptable. We have updated Chapter 7 to include reference to our most recent financial assurance letter dated July 19, 1990. We suggest that changes to Chapter 7 of the license application to reflect the updated decommissioning funding plan be made after its submittal.

Page Comment

8-1 Chapter 8 should be revised to reflect the changes authorized in Amendment 19 and to conform to the regulatory requirements in 10 CFR 70.22.

Response:

Page 8-1 of the Renewal Application has been revised to reflect the changes authorized in Amendment 19.

The Emergency Plan submitted April 6, 1992, has not yet been approved by the NRC. When the Emergency plan was submitted, we indicated the Plan would be implemented within 180 days after NRC approval. It is therefore our intent to submit an amendment to the license reflecting implementation of the Plan within 180 days after NRC approval.

10-3 In Section 10.2.6, provide a map indicating the locations of the chemical materials which are stored onsite. For each chemical material, indicate the maximum capacity of the onsite storage.

Response:

Section 10.2.6 has been revised to indicate the quantity of each significant chemical stored on the Hematite site. Figure 10-3, "Chemical and Other Hazardous Materials Storage Locations" has been added. Subsequent figures were re-numbered to reflect this addition.

10-4 Section 10.3 should indicate the locations (stack, process areas, or equipment) where the air cleaning equipment, as described in Section 3.2.2, is being used.

Response:

Section 10.3 has been revised to indicate the locations where air cleaning equipment is used.

10-5 Section 1.6 should address the use of radioactive contaminated calcium fluoride and limestone as fill material.

Response:

Section 10.4.2 has been revised to delete the discussion on the use of spent limestone as clean fill material on the Hematite site. Section 1.6 has been revised to discuss the use of spent limestone as fill material. Any material

Page Comment

used for fill will be surveyed to demonstrate contamination levels are less than 30 picocuries/gram.

11-1 In Section 11.1, the management program for assessing the criticality safety program should be described.

The audit and inspection program should be described in detail. The description should include the following: (1) responsibilities of staff positions and committees; (2) reporting levels; (3) corrective action program including responsibilities for designating actions, determining sufficiency of actions, tracking actions, and performing root-cause analysis; (4) methods established for observing operations to verify that the conditions and assumptions used in the safety analyses are valid and are controlled by operating procedures and design documents; and (5) review of engineered controls by evaluation of programs established for maintenance, surveillance, and functional testing.

Response:

The response to this question will be provided in the second submittal.

11-3 Update the names and resumes of key personnel.

Response:

Section 11.3, "Education and Experience of Key Personnel", has been updated to reflect organizational changes since the initial submittal of the Renewal Application.

11-18
A. In Section 11.4, the management control program for procedures should be discussed and the methods and practices for development, revision, review, approval, and implementation of written procedures for plant operations, including maintenance and surveillance, should be described. The discussion should include the periodic review used to ensure continued applicability and adequacy of procedures and the responsibilities for updating procedures.

Response:

The response to this question will be provided in the second submittal.

Page Comment

B. In Section 11.5, the program for nuclear criticality safety (NCS) training, including the responsibilities of the NCS staff, should be described. The following should be discussed: personnel responsible for content of NCS training, responsibilities for evaluating NCS training program, guidance to aid supervision in conducting on-the-job training; training requirements for supervision, maintenance personnel, engineering, and management; and the system for maintaining training records.

Response:

The response to this question will be provided in the second submittal.

C. Section 2.5 of the license application states that training is supplemented by regularly scheduled meetings conducted by line supervision and specialists. "Regularly" should be defined and any guidance provided to supervision for conducting the meetings should be identified.

Response:

Section 2.5, "Training", has been revised to indicate on-the-job training is supplemented by specialized training in various safety topics. The term "Regularly" has been deleted since it is not defined. Although the specialized training is conducted frequently, it would not be correct to state that it is performed on a monthly, quarterly, or annual basis.

D. In Section 11.6, the configuration control program should be described. The administrative control program and procedures instituted for keeping design basis documentation current should be discussed. (See configuration control discussion in Attachment 2.)

Response:

The response to this question will be provided in the second submittal.

Page Comment

12-1

A. Section 12.1 should describe the administrative procedures for implementing the ALARA policy and for issuing the RWPs.

Response:

Section 12.1 has been revised to include the ALARA program and to list the procedures which are used to implement it. The Special Evaluation Traveler, which is used as an RWP, is described in Section 3.1.2.

B. Section 12.1 should contain a list of health and safety procedures that are being used for the health physics program.

Response:

Section 12.1 has been revised to include a list of many of the health and safety procedures which are in place.

C. Section 12.3 should describe the monitoring program for verifying that the shallow dose equivalent received by personnel handling uranium material meets the provisions in 10 CFR 20.101(a).

Response:

The response to this question will be provided in the second submittal.

12-2

A. Expand Section 12.4 to describe how the radiological survey will be performed. Indicate the instruments or equipment used in conducting measurements for external radiation dose rates, airborne radioactivity concentrations, surface contamination, protective clothing contamination, and personnel contamination.

Response:

The response to this question will be provided in the second submittal.

Page Comment

B. Section 12-4 should contain a plant layout identifying the contaminated areas and their exit point(s) where radiation monitoring is provided.

Response:

The response to this question will be provided in the second submittal.

12-3 Section 12.10.1 should provide the personnel exposure results including shallow-dose equivalent for 1989 through 1991. Specific dose ranges above 0.5 Rem should be included.

Response:

Section 12.10.1 has been updated to include more recent external exposure data for Hematite workers. Shallow-dose equivalent information has not been provided, however, since it is not readily available in a reduced form. Film badge data is reviewed each month and transposed to the individual workers records. This information is available for review during the periodic NRC inspections.

12-4 In Section 12.10.2, provide internal exposure records for 1989-1991. Provide weekly internal exposure records (mpc-hr) for the past 3 years for those workers who handled the soluble uranium material.

Response:

Section 12.10.2 has been updated to include more recent internal radiation exposure data for Hematite workers.

We have not provided weekly internal exposure records for workers handling soluble uranium materials because we do not have any process areas in which the workers are strictly dedicated to those activities and measured for soluble exposures. Internal exposure data for workers is tracked in reference to insoluble uranium exposure limits, which are more limiting than soluble limits.

<u>Page</u> <u>Comment</u>

12-5 Describe specific changes to equipment and procedures to reduce airborne exposures. Provide data showing the effectiveness of the changes.

Response:

Changes to equipment and procedures have not been described in Part II of the license application since the Part II Safety Demonstration is meant to describe current processes. To describe changes with respect to the past or planned changes for the future would confuse the Safety Demonstration. In lieu of revising the license application we offer the following description in response to this comment:

Extensive improvements have been made to the ventilation systems at the Hematite facility, and the improvements are still ongoing. Older facilities are in the process of being upgraded with new HEPA filter systems. The new pellet plant in Building 254 includes numerous process improvements which results in reduced exposures. A new enclosure has been installed over the pellet press to reduce airborne exposure. In general, systems which could result in airborne contamination are modified to enclose them, such as the use of large closed blenders for UO_2 powder mixing, in lieu of small open containers. In addition, the centrifuge in the wet recovery area has been improved by enclosing its discharge.

While the above improvements are designed to reduce airborne exposures, data showing the effectiveness of the changes has not been collected. There is little advantage in collecting such data.

12-6
A. Describe the methodology used in assessing personnel internal exposure levels. Include bioassay (in-vivo and in-vitro measurements) and airborne sampling results.

Response:

The response to this question will be provided in the second submittal.

Page Comment

B. Section 12.12 should contain a quality assurance program for in-vitro and in-vivo measurements performed by the licensee and vendors.

Response:

The response to this question will be provided in the second submittal.

C. Section 12.13 should describe the following:

(1) The method(s) used in determining the locations of the breathing zone air samplers for their representativeness.

(2) The airborne radioactive concentration level that would require shutting down the operations.

Response:

The response to this question will be provided in the second submittal.

14-1

A. In Section 14.1, the Nuclear Fuel Manufacturing Program (NFMP) documentation system describes administrative and technical procedures related to NCS. The NFMP should be endorsed in Chapter I. The administrative and technical procedures should be identified in this section.

Response:

The response to this question will be provided in the second submittal.

B. Management control programs addressing the establishment and implementation of design basis documentation, process safety analyses, operating procedures, training, configuration control, incident investigations, audits, maintenance and surveillance should be described. This section should discuss information pertaining to the programmatic framework for administrative and procedural controls and the organizational framework that allows the staff to implement programs. Enough detail should be provided to allow assessment of the organizational and programmatic structure to justify reliance on administrative and operational controls. Methods of implementation should be described.

Page Comment

Response:

The response to this question will be provided in the second submittal.

14-7 In Section 14.3.2, the accident analysis process should be outlined.

Response:

The response to this question will be provided in the second submittal.

14-16 In Section 14.4, the maintenance for borosilicate glass raschig rings should be described.

Response:

Section 14.4, "Fixed Poisons", has been revised to update the requirements for maintenance of borosilicate glass raschig rings. Regulatory Guide 3.1, Revision 2 has been referenced.

14-19

A. In Section 14.6.2, discuss the written report that demonstrates the validation of analytical methods and establishes the range of applicability and biases.

Response:

The response to this question will be provided in the second submittal.

B. In Section 14.7, special controls used to ensure nuclear safety should be described.

(1) For fissile transfers to unfavorable geometry vessels, a physical barrier should prevent the inadvertent transfer of fissile solutions. Solution transfers should be limited so that vessels never contain more than a fraction of the calculated minimum critical mass. Uranium concentration should be limited by controlled and verified chemical characteristics of the materials involved. Two independent methods for determining concentration should be provided to confirm that the limit is satisfied. Uranium concentration should be limited by on-line measurement, and if the limit is reached, automatic controls should prevent continued release.

Page Comment

(2) Techniques used for the investigation of SNM accumulations and for safe removal of any accumulated material should be described. Procedures should include the components to be inspected, specific action levels, inspection frequencies, and response actions.

(3) The program which demonstrates that equipment and instrumentation used for measuring process variables meets safety and design criteria discussed in the safety analyses should be described.

Instruments used to detect process conditions and the systems used to control processes should be discussed, including testability, redundancy, and failure conditions. Process instruments that are used to sense and control parameters should be discussed and the mechanism employed to ensure the proper functioning condition, e.g., functional testing and calibration, should be described.

The program that ensures that adequate sampling measurements, control instrumentation, and safety monitoring capabilities are provided and maintained operational should be described. Provisions for obtaining samples for process analysis and controls necessary to ensure that operations are within prescribed limits should be discussed. The facilities and analytical equipment used to perform the analyses should be described. The laboratory analyses that provide confirmation of process conditions should be described.

The program for improving the analytical methods and measurements for maintaining such a program and for incorporating improvements into the analytical and measurements methods should be described.

Systems should be designed so that when sampling is part of a control, representative sampling may be obtained.

Response:

The response to this question will be provided in the second submittal.

15-1 In Section 15.1, the dry powder criterion relies on moisture analysis. The measurement techniques employed should be described, and the technical basis for their validity should be discussed. The limits and technical basis for UO2 additives and moisture control should be discussed. The controls that designate moderation control areas should be discussed. The controls preventing pneumatic transfer of moderators should be described.

Page Comment

Response:

Section 15.1, "Process Outline and Moderation Control", has been updated to include additional detail regarding moderation control.

15-2 In Section 15.2.1, the controls used to verify the isotopic contents of cylinders should be described.

Response:

Section 15.2.1 has been revised to describe our use of DOE or other independent test results for verification of isotopic content.

15-3 In Section 15.2.2.1, the "valving arrangement" that prevents the interconnection of two cylinders should be described. The dimensions of the steam chamber, condensate drain line, and piping insulation should be provided.

Response:

Section 15.2.2.1 has been updated to include a description of the valving arrangement between the two cylinders. Dimensions have also been provided in this section.

15-5

A. The nuclear safety of the steam chamber should be analyzed for uranyl fluoride solution.

Response:

Section 15.2.2.1, under "Nuclear Safety", has been updated to include a discussion of the safety analysis for a scenario in which UF_6 leaks into the steam chamber.

B. The controls used to prevent backflow of moderating materials from conversion lines to UF6 cylinders should be described.

Response:

Section 15.2.2.1, under "Safety Features", has been revised to include a discussion of the controls for preventing the backflow of moderating materials to the UF_6 cylinders.

Page Comment

15-23 In Section 15.3.4, indicate that batch control uses an interlock.

Response:

Section 15.3.5 has been revised to indicate that the criticality safety of the entire system is independent of batch control. Analysis has been performed and documented in Section 15.3.5.4 which demonstrates the system can be full of UO_2 without presenting a criticality concern.

Enclosure II to
ML-92-045

COMBUSTION ENGINEERING, INC.

HEMATITE NUCLEAR FUEL MANUFACTURING FACILITY

RESPONSE TO NRC QUESTIONS ON THE

LICENSE RENEWAL APPLICATION

LIST OF AFFECTED PAGES

October 1992

COMBUSTION ENGINEERING, INC.

HEMATITE NUCLEAR FUEL MANUFACTURING FACILITY

RESPONSE TO NRC QUESTIONS ON THE

LICENSE RENEWAL APPLICATION

LIST OF AFFECTED PAGES

Combustion Engineering, Inc. is responding to an NRC request for additional information on our Hematite License Renewal Application dated November 22, 1989, supplemented June 17, 1991. Several responses to the NRC requests have resulted in changes being made to the existing license renewal application. The following identifies the changed license pages. The affected pages are provided as change pages in Enclosure III.

The license application pages affected are as follows:

Delete Page			Add Page		
Page No.	Rev.	Date	Page No.	Rev.	Date
PART I			PART I		
Chapter 1			Chapter 1		
1-1	0	November 22, 1989	1-1	0	October 2, 1992
1-2	0	November 22, 1989	1-2	0	October 2, 1992
1-3	0	November 22, 1989	1-3	0	October 2, 1992
-	-	-	1-4	0	October 2, 1992
Chapter 2			Chapter 2		
2-1	0	November 22, 1989	2-1	0	October 2, 1992
2-2	0	November 22, 1989	2-2	0	October 2, 1992
2-3	0	November 22, 1989	2-3	0	October 2, 1992
2-4	0	November 22, 1989	2-4	0	October 2, 1992
2-5	0	November 22, 1989	2-5	0	October 2, 1992
2-6	0	November 22, 1989	2-6	0	October 2, 1992
2-7	0	November 22, 1989	2-7	0	October 2, 1992
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-	-	-	2-9	0	October 2, 1992
-	-	-	2-10	0	October 2, 1992

List of Affected Pages - continued

Delete Page Page No.	Rev.	Date	Add Page Page No.	Rev.	Date
Chapter 3			Chapter 3		
3-1	0	November 22, 1989	3-1	0	October 2, 1992
3-2	0	November 22, 1989	3-2	0	October 2, 1992
3-3	0	November 22, 1989	3-3	0	October 2, 1992
3-4	0	November 22, 1989	3-4	0	October 2, 1992
3-5	0	November 22, 1989	3-5	0	October 2, 1992
3-6	0	November 22, 1989	3-6	0	October 2, 1992
-	-	-	3-7	0	October 2, 1992
Chapter 4			Chapter 4		
4-1	0	June 17, 1991	4-1	0	October 2, 1992
4-2	0	June 17, 1991	4-2	0	October 2, 1992
4-3	0	June 17, 1991	4-3	0	October 2, 1992
4-4	0	June 17, 1991	4-4	0	October 2, 1992
4-5	0	June 17, 1991	4-5	0	October 2, 1992
4-6	0	June 17, 1991	4-6	0	October 2, 1992
4-7	0	June 17, 1991	4-7	0	October 2, 1992
4-8	0	June 17, 1991	4-8	0	October 2, 1992
4-9	0	June 17, 1991	4-9	0	October 2, 1992
4-10	0	June 17, 1991	4-10	0	October 2, 1992
4-11	0	June 17, 1991	4-11	0	October 2, 1992
-	-	-	4-12	0	October 2, 1992
-	-	-	4-13	0	October 2, 1992
-	-	-	4-14	0	October 2, 1992
Chapter 5			Chapter 5		
5-1	0	October 11, 1991	5-1	0	October 2, 1992
Chapter 7			Chapter 7		
7-1	0	November 22, 1989	7-1	0	October 2, 1992
Chapter 8			Chapter 8		
8-1	0	November 22, 1989	8-1	0	October 2, 1992

List of Affected Pages - continued

Delete Page			Add Page		
Page No.	Rev.	Date	Page No.	Rev.	Date
PART II			PART II		
Chapter 10			Chapter 10		
10-1	0	November 22, 1989	10-1	0	October 2, 1992
10-2	0	November 22, 1989	10-2	0	October 2, 1992
10-3	0	November 22, 1989	10-3	0	October 2, 1992
10-4	0	November 22, 1989	10-4	0	October 2, 1992
10-5	0	November 22, 1989	10-5	0	October 2, 1992
10-6	0	November 22, 1989	10-6	0	October 2, 1992
10-7	0	November 22, 1989	10-7	0	October 2, 1992
10-8	0	November 22, 1989	10-8	0	October 2, 1992
10-9	0	November 22, 1989	10-9	0	October 2, 1992
10-10	0	November 22, 1989	10-10	0	October 2, 1992
10-11	0	November 22, 1989	10-11	0	October 2, 1992
10-12	0	November 22, 1989	10-12	0	October 2, 1992
-	-	-	10-13	0	October 2, 1992
-	-	-	10-14	0	October 2, 1992
Chapter 11			Chapter 11		
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11-4	0	November 22, 1989	11-4	0	October 2, 1992
11-5	0	November 22, 1989	11-5	0	October 2, 1992
11-6	0	November 22, 1989	11-6	0	October 2, 1992
11-7	0	November 22, 1989	11-7	0	October 2, 1992
11-8	0	November 22, 1989	11-8	0	October 2, 1992
11-9	0	November 22, 1989	11-9	0	October 2, 1992
11-10	0	November 22, 1989	11-10	0	October 2, 1992
11-11	0	November 22, 1989	11-11	0	October 2, 1992
11-12	0	November 22, 1989	11-12	0	October 2, 1992
11-13	0	November 22, 1989	11-13	0	October 2, 1992
11-14	0	November 22, 1989	11-14	0	October 2, 1992
11-15	0	November 22, 1989	-	-	-
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11-17	0	November 22, 1989	-	-	-
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List of Affected Pages - continued

Delete Page Page No.	Rev.	Date	Add Page Page No.	Rev.	Date
Chapter 12			Chapter 12		
12-1	0	November 22, 1989	12-1	0	October 2, 1992
12-2	0	November 22, 1989	12-2	0	October 2, 1992
12-3	0	November 22, 1989	12-3	0	October 2, 1992
12-4	0	November 22, 1989	12-4	0	October 2, 1992
12-5	0	November 22, 1989	12-5	0	October 2, 1992
12-6	0	November 22, 1989	12-6	0	October 2, 1992
-	-	-	12-7	0	October 2, 1992
Chapter 14			Chapter 14		
14-22	0	June 17, 1992	14-22	0	October 2, 1992
Chapter 15			Chapter 15		
15-2	0	June 17, 1991	15-2	0	October 2, 1992
15-3	0	June 17, 1991	15-3	0	October 2, 1992
15-4	0	June 17, 1991	15-4	0	October 2, 1992
15-5	0	June 17, 1991	15-5	0	October 2, 1992
-	-	-	15-5a	0	October 2, 1992
-	-	-	15-5b	0	October 2, 1992
15-23	0	June 17, 1991	15-23	0	October 2, 1992
15-48	0	June 17, 1991	15-48	0	October 2, 1992
15-49	0	June 17, 1991	15-49	0	October 2, 1992
15-50	0	June 17, 1991	15-50	0	October 2, 1992
15-50a	0	June 17, 1991	15-50a	0	October 2, 1992

Enclosure III to
ML-92-045

COMBUSTION ENGINEERING, INC.

HEMATITE NUCLEAR FUEL MANUFACTURING FACILITY

RESPONSE TO NRC QUESTIONS ON THE

LICENSE RENEWAL APPLICATION

AFFECTED PAGES

October 1992

CHAPTER 1 STANDARD CONDITIONS AND SPECIAL AUTHORIZATIONS

1.1 Name, Address and Corporate Information

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

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

1.2 Site Location

The Hematite fuel manufacturing facility of Combustion Engineering, Inc. is located on a site of about 212 acres in Jefferson County, Missouri, approximately 3/4 mile northeast of the unincorporated town of Hematite, Missouri and 31 miles south of the city of St. Louis, Missouri. Activities with special nuclear materials are conducted within an 8 acre, controlled access area near the center of the site and adjacent to the access road, Highway P. Nuclear fuel manufacturing activities occur within the fenced, controlled area. These activities include conversion of UF_6 to UO_2, fabrication of UO_2 nuclear fuel pellets, and related processes.

1.3 License Number and Period of License

This application is for renewal of Special Nuclear Material License No. SNM-33 (NRC Docket 70-36). Renewal is requested to cover a period of ten (10) years.

1.4 Possession Limits

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

Material	Form	Quantity
Uranium enriched to maximum of 5.0 weight percent in the U-235 isotope	Any (Excluding metal powders)	8,000 kilograms contained U-235
Uranium to any enrichment in the U-235 isotope	Any (Excluding metal powders)	350 grams
Source material (Uranium and Thorium)	Any (Excluding metal powders)	50,000 kilograms
Cobalt 60	Sealed sources	40 millicuries total
Mixed Activation and Fission Product Calibration Sources Including Am-241	Solid Sources	200 microcuries total

1.5 Authorized Activities

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

Number	Name	Present Utilization
101	Tile Barn	Emergency Center and equipment storage
110	Office Building	Guard Station and Offices
120	Wood Barn	Equipment storage
-	Oxide Building and Dock	UF_6 to UO_2 Conversion, UF_6 receiving
235	West Vault	Source material storage
240	240-1	Offices and Cafeteria
	240-2	Recycle and Recovery area
	240-3	Incinerator, SNM storage and waste processing
	240-4	Laboratory and Maintenance Shop
252	South Vault	Radioactive Waste Storage
253	Utility Building	Steam Supply, SNM Storage, Operating Supplies, Offices, and Liquid Waste Solidification
254	254-1	UO2 storage, blending, and pressing
	254-2	UO2 oxidation, reduction, dewaxing, and sintering
	254-3	UO2 grinding and pellet packaging
255	Pellet Plant	Pellet fabrication, storage, and packaging
256	Warehouse	Shipping, Receiving and Storage

1.6 Exemptions and Special Authorizations

The following are specific exemptions and special authorizations of this license application:

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

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

(c) Release calcium fluoride (spent limestone) from the conversion process dry scrubbers for use as fill materials on site, providing the uranium alpha activity is less than 30 picocuries per gram.

(d) At 2 year intervals from the date of NRC approval of this renewal application, the licensee shall update the demonstration sections of the renewal application to reflect the licensee's current operations. The updates to the application shall, as a minimum, include information for the health and safety section of the application as required by 10 CFR 70.22(a) through 70.22(f) and 70.22(i) and operational data and information on environmental releases as required by 70.21. In lieu of an update at the end of the 10-year renewal period, the licensee shall file a renewal application on or before ten years from the date of NRC approval of this renewal application.

CHAPTER 2 ORGANIZATION AND ADMINISTRATION

2.1 Organizational Responsibilities and Authority

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

2.1.1 Plant Manager, Hematite

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

2.1.2 Manager, Nuclear Licensing, Safety and Accountability

The Manager, Nuclear Licensing, Safety and Accountability (NLS&A) reports to the Plant Manager. He manages radiological protection and industrial safety, SNM accountability, criticality safety, licensing, emergency planning, and environmental protection. His activities include review and approval of procedures for control, sampling, measurement and physical inventory of SNM, and auditing of plant operations. He is responsible for ensuring workers are provided radiological, criticality and industrial safety training. He reviews results from personnel and environmental monitoring and facility activities to ensure compliance with the requirements of License No. SNM-33. To enforce compliance, he has authority to halt any operation at the Hematite facility, and the operation shall not restart until approved by the Plant Manager or a duly authorized alternate.

2.1.3 Superintendent, Production

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

2.1.4 Manager, Engineering

The Manager, Engineering reports to the Plant Manager. He manages the engineering of new equipment and of modifications to existing equipment. With support from his staff, his activities include recommendation, development and qualification of manufacturing processes, specification of process control methods and design, procurement and installation of processing equipment.

2.1.5 Nuclear Criticality Specialist

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

2.1.6 Supervisor, Health Physics

The Supervisor of Health Physics reports to the Manager of Nuclear Licensing, Safety and Accountability. He supervises the health physics technicians in the radiological surveillance of activities that involve radioactive materials, in personnel radiation monitoring and in the collection and measurement of environmental samples. He has the authority to suspend unsafe operations.

2.1.7 Health Physics Specialist

The Health Physics Specialist reports to the Manager of Nuclear Licensing, Safety and Accountability. His activities include observation of plant operations and evaluation of results from personnel and environmental monitoring. He compares quantitative measurements and other observations of facility activities with the requirements of License No. SNM-33.

2.1.8 Health Physics Technicians

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

2.2 Personnel Education and Experience Requirements

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

2.3 Hematite Plant Safety Committee

The Hematite Plant Safety Committee meets at least once each calendar quarter to review plant operations, to compare them with selected safety requirements of Part I and the License Conditions and to consider other aspects of safety the Committee believes appropriate. The Plant Safety Committee shall perform an annual review of each of the following:

- o Environmental protection trends
- o Radiation safety trends
- o Criticality safety practices
- o Industrial safety trends
- o Adequacy of emergency planning and drills
- o Effectiveness of ALARA program
- o Internal inspection and audit reports
- o Abnormal occurrences and accidents including recommendations to prevent recurrence
- o Review of significant physical facility changes in the pellet shop and significant changes to operations involving radiation and/or nuclear criticality safety

The review of findings and recommendations of corrective action shall be reported to the Plant Manager for action.

The Committee Chairman or Plant Manager determines which committee members, as a minimum, shall attend each quarterly meeting, according to the topics to be considered. The Committee submits a quarterly meeting report to the Hematite manager level personnel and the Plant Manager at Hematite. The Plant Manager appoints the committee members to represent, as a minimum, engineering, production, health physics, and criticality safety. He may also approve alternate(s) for the members.

Minimum education and experience requirements for the Chairman are in Table 2-1. The Committee is composed of senior personnel from the technical staff of Combustion Engineering's Nuclear Power organization who have at least five (5) years experience in the nuclear industry. The Committee Chairman or Plant Manager may invite participation by others from within Hematite or from the staff at Windsor.

2.4 Approval Authority for Personnel Selection

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

2.5 Training

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

The training and personnel safety program continues with on-the-job training supplemented by training in specialized topics such as personnel protective equipment, industrial safety, accident prevention, and other safety topics. Production Supervisors receive formal training in radiation and criticality control. Testing determines when they have sufficient knowledge to enable them to carry out their training functions. Operating personnel receive a re-training course in criticality control and radiation safety on a biennial basis. The effectiveness of retraining is determined by testing. Formal training shall be documented. The health physics technicians will receive professional related training at least biennially.

2.6 Operating Procedures

Operations which involve licensed material shall be conducted in accordance with approved written procedures. Operating Procedures, called Operation Sheets, are issued and controlled by Quality Control. They provide the detailed instructions for equipment operation and material handling and the limits and controls required by the License. Operation Sheets are the basic control document; before issuance or revision they require signed approval by the Managers of Engineering, Production, Quality Control, and Nuclear Licensing, Safety, and Accountability. In the Manager's absence, another individual meeting the Manager's minimum education and experience requirements, or the Plant Manager, may provide approval. Health Physics activities will be conducted in accordance with approved written procedures; these procedures must be approved by the Manager, NLS&A.

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

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

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

The NLS&A Manager shall grant approval only when:

a. A nuclear criticality safety evaluation has been performed based on the criteria and standards of Chapters 3 and 4 by a person who meets the education and experience requirements for a Nuclear Criticality Specialist (and who may be the NLS&A Manager). This evaluation shall be in sufficient detail to permit subsequent review.

b. The criticality safety evaluation has been reviewed by a person who fulfills the education and experience requirements for a Nuclear Criticality Specialist (and who may be the NLS&A Manager). This individual will be different from the person who performed the evaluation. This review is based on the criteria and standards of Chapter 4 and includes verification of each of the following:

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

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

Review and verification shall include written approval by the reviewer.

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

2.7 Audits and Inspections

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

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

Planned and documented quarterly inspections, performed by an individual who meets the education and experience requirements of the NLS&A Manager, cover criticality control, radiation safety and industrial safety. The inspection of criticality control shall be performed by an individual meeting at least the education and experience requirements of a Nuclear Criticality Specialist and at least one of the quarterly inspections per year regarding criticality control will be by an individual who is not the NLS&A Manager. Items requiring corrective action are documented in a report distributed to the Plant Manager and manager level staff. The Superintendent, Production, is responsible for corrective action, except where another manager is specifically designated. Follow-up actions taken by the Superintendent, Production, or responsible manager, shall be documented. Documentation shall be maintained for at least the period stated in Section 2.9.

Annual audits are conducted in which the results of previous inspections or audits are reviewed, as an evaluation of the effectiveness of the program. These audits may also involve a detailed review of non-safety documents such as operation procedures, shop travelers, etc., and are documented by a formal report to the President, Nuclear Fuel. Annual audits are performed by a team appointed by the President, Nuclear Fuel. Personnel on the team will not have direct responsibility for the function and areas being audited. The team shall include, as a minimum, a Nuclear Criticality Specialist and a radiation specialist who shall audit criticality and radiation safety, respectively. The radiation specialist who conducts the annual audit shall have as a minimum, a Bachelor's degree in Science or Engineering with two years experience in operating health physics for uranium bioassay techniques, internal exposure controls and radiation measurement techniques. The annual audit will review ALARA requirements in conformance with Regulatory Guide 8.10, Revision 1-R, dated May 1977, as applicable. The NLS&A Manager shall be responsible for follow-up of recommendations made by the audit team.

2.8 Investigations and Reporting

Events specified by applicable regulations or license conditions shall be investigated and reported to NRC. The NLS&A Manager shall be responsible for conducting the investigation and documentation of reportable events.

Non-reportable occurrences shall be investigated and documented as appropriate. Such reports shall be available for NRC inspection.

2.9 Records

Retention of records required to be maintained by the regulations, and by the conditions of this license, shall be the responsibility of the cognizant manager. Records of NLS&A evaluations and approvals shall be retained for a period of at least six months after use of the operation has been terminated, or for two years, whichever is longer. Other safety significant records shall be retained for at least two years.

TABLE 2-1

MINIMUM EDUCATION AND EXPERIENCE REQUIREMENTS FOR KEY PERSONNEL

POSITION Described In Section No.	Title	Education	Experience (Years/Field)
2.1.1	Plant Manager	Bachelors, Science or Engineering	5/Nuclear manufacturing
2.1.2	Manager, NLS&A	Bachelors, Science or Engineering	5/Health Physics with 2/Operational health physics with uranium bioassay techniques, internal exposure control, and radiation measurement techniques
2.1.3	Superintendent, Production	Bachelors, Science, Engineering or Manufacturing	2/Nuclear manufacturing industry
2.1.4	Manager, Engineering	Bachelors, Science or Engineering	5/Engineering design or process, systems or facilities
2.1.5	Nuclear Criticality Specialist	Bachelors, Science or Engineering	2/Nuclear criticality evaluations.
2.1.6	Supervisor, Health Physics	High School Diploma	5 Total/Nuclear industry, with 3/Senior Health Physics Technician
2.1.7	Health Physics Specialist	Bachelors, Science or Engineering	2/Operational Health Physics applicable to fuel manufacturing
2.1.8	Health Physics Technician	High School Diploma or GED Equivalent	6 months/Training and experience in radiation protection activities
2.3	Chairman, Plant Safety Committee	Bachelors, Science or Engineering	5/Nuclear manufacturing industry

CHAPTER 3 RADIATION PROTECTION

3.1 Special Administrative Requirements

3.1.1 ALARA Policy

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

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

3.1.2 Radiation Work Permit Procedures

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

3.2 Technical Requirements

3.2.1 Restricted Areas - Personnel Contamination Control

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

3.2.2 Ventilation

Air flow shall be from areas of lower to areas of higher contamination. Hoods, glove boxes, or local exhaust will be used to control contamination and airborne concentrations. All dispersible forms of uranium will be handled in ventilated enclosures having sufficient air flow to assure minimum face velocities of 100 Fpm. Face velocities will be checked weekly by NLS&A, except during periods when not in use. This procedure checks the entire ventilation system, including the HEPA filters. HEPA filter and pre-filter banks are provided with differential pressure gauges for diagnostic purposes. Filters/prefilters are normally changed if the differential pressure across the filter exceeds six inches of water. Ventilation systems with DOP ports shall be DOP tested in place after any disturbance of the HEPA filters. New HEPA systems will be equipped with the DOP testing provision.

The direction of air flow in the process buildings shall be checked at least annually and documented. Major design changes having a potential to impact air flow direction may require a re-check of the air flow direction once the design change has been completed. If the air flow direction is not acceptable, action will be taken.

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

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

a. Cyclone Collectors

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

b. High Efficiency Particulate Air Filters

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

c. Wet Scrubbers

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

d. Dry Scrubbers

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

e. Fabric Filters

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

f. Special Filters

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

3.2.3 Work - Area Air Sampling

3.2.3.1 Air Sampling Criteria

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

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

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

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

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

3.2.3.2 Airborne Concentrations

a. Airborne levels in excess of 25% of the maximum permissible concentration shall require posting in accordance with 10 CFR 20 and an investigation of the causes.

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

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

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

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

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

3.2.4 Radioactivity Measurement Instruments

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

a. Nuclear Alarm System

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

b. Alpha Counting System

Minimum detectability shall be 10 dpm.

c. Alpha Survey Meters

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

d. Air Sampling Equipment

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

e. Beta-Gamma Survey Meters

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

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

f. Beta-Gamma Counting System

Minimum detectability shall be 200 dpm

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

3.2.5 Radiation Exposures

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

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

3.2.6 Surface Contamination

3.2.6.1 Special Surveys

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

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

3.2.6.2 Routine Surveillance

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

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

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

3.2.7 Bioassay

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

3.2.8 Respiratory Protection

The respiratory protection program shall be conducted in accordance with Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection". Protection factors are used in estimating exposure to individuals using respirator equipment.

CHAPTER 4 NUCLEAR CRITICALITY SAFETY

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

Administrative conditions define:

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

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

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

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

4.1 Administrative Conditions

4.1.1 Process Design Philosophy

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

a) Process design, in so far as the handling and storage of SNM, shall incorporate sufficient factors of safety such that at least two unlikely, independent, and concurrent changes in process conditions are required before a criticality accident may occur. Process design which does not meet this requirement shall be explicitly approved in Chapter 1, Section 1.6, of this application.

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

c) Before a new operation with SNM is begun or an existing operation is changed, it shall be determined that the entire process will be subcritical under normal and operating conditions, consistent with paragraph a) of this section and applicable technical criteria of Section 4.2.1.3.

4.1.2 Positions Responsible for Criticality Safety

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

Proposed changes or modifications to SNM processing, handling, or storage equipment or related operations shall be reviewed for criticality safety. Manufacturing Engineering (or some other specified functional group) shall process change requests and secure the necessary management and safety reviews and approvals prior to implementation of the change. Significant changes, as determined by the NLS&A Manager, to operations involving radiological and/or criticality safety are also reviewed by the Hematite Plant Safety Committee. Facility change requests requiring a criticality safety review shall be evaluated by a Nuclear Criticality Specialist.

4.1.3 Documenting Criticality Evaluations and Reviews

Criticality evaluations associated with facility changes affecting the handling and storage of SNM in Nuclear Manufacturing shall be documented by the Nuclear Criticality Analyst and independently reviewed.

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

These limits and controls shall be incorporated into applicable written procedures and postings and approved by a qualified Nuclear Criticality Safety Specialist and the Manager, NLS&A. Production and line supervisory personnel shall assist in the preparation of written procedures and postings. Day-to-day monitoring of workers for conformance to criticality limits and controls and administrative procedures is carried out by line supervision and Health Physics technicians.

Documentation of the criticality evaluations by the Criticality Specialist/Analyst shall be sufficiently detailed such that an independent reviewer can reconstruct the analysis and bases for the conditions presented. Criticality evaluations shall include assumptions affecting criticality safety process limits and controls. If explicit analyses using validated methodologies are employed, the margin to criticality and a clear definition of assumed off-nominal conditions shall be provided.

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

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

4.1.4 Written Procedures

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

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

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

c) Special Evaluation Traveler - The Special Evaluation Traveler (SET) is employed for those jobs involving the handling and/or storage of SNM which are not covered by standard procedures. A SET may supplant operation sheets in a development and testing environment.

4.1.5 Posting of Limits and Controls

Work and storage areas where SNM is handled, processed, or stored shall be posted with the nuclear safety limits and controls applicable to each area and approved by the Manager, NLS&A and a Nuclear Criticality Specialist. NLS&A shall maintain a current record of: 1) the review and approval of each posting, 2) the location of each posting, and 3) the content of each posting.

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

4.1.6 Labeling of Special Nuclear Material

Mass-limited containers employed in the handling or storage of special nuclear material shall be labeled as to their contents. If SNM is in the container, the amount, enrichment and type shall be indicated; if empty, the container shall be so labeled or placed in designated areas for empty containers. Uncovered empty containers do not require an empty sign.

4.1.7 Preoperational Testing and Inspection

Prior to startup of a new or modified process, an inspection of equipment, procedures, and postings shall be carried out by Hematite Engineering and NLS&A representatives to assure completeness and consistency between safety evaluations, equipment design, installation, written procedures, and postings. This inspection shall be documented as part of the records for this facility change and retained according to internal procedures and Section 2.9 of this license.

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.

4.1.8 Criticality Safety Design

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

4.2 Technical Criteria

4.2.1 Individual Units

4.2.1.1 Safe Individual Units (SIU)

Minimum critical values of safety parameters shall be based on 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 shall be reduced by the following safety margins.

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

For SIUs determined form calculated data, the calculations shall be performed using validated computer analysis methods.

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

4.2.1.2 Subcritical Units (Subcrits)

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

The configuration and composition of these subcritical units may depend upon the process involved. Criticality safety is assured through defined limits and controls. These limits and controls may include allowed individual SNM unit geometries which are less conservative than safe geometry, defined configurations of individual SNM units in a given process layout, engineered safeguards where necessary, and administrative controls in the form of written and approved instructions sheets and postings.

4.2.1.3 Criteria

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

(b) Nuclear safety evaluations shall include credible sources of internal moderation.

(c) Criticality safety evaluations shall 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.

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

(e) Reactivity hold-down by other than fixed poisons shall not be employed in criticality evaluations. Borosilicate Glass Raschig Rings may be employed in solutions of fissile material in a manner consistent with ANSI/ANS 8.5-1986. The effect of structural parasitics, either normal or enhanced, shall be evaluated in a manner which examines both elastic and inelastic scattering contributions to the multiplication factor. Use of enhanced structural parasitics, e.g., boron stainless steel, shall be contingent upon a program to periodically verify the presence of the parasitic additive.

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

(g) Computer analysis methods shall be validated in accordance with the criteria of Section 4.2.3.2 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 shall be less than or equal to 0.95 including applicable biases and calculational uncertainties.

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

4.2.2 Multiple Units and Arrays

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

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

4.2.2.1 Spacing of Safe Units

The following criteria shall be employed:

(a) Application of the surface density method of spacing safe mass, volume, or cylinder diameter limited units requires meeting the following criteria:

  (1) Safe mass, volume, or cylinder diameter limited units shall meet the maximum values defined in Table 4-1.

  (2) The spacing areas for the safe mass, volume, or cylinder diameter limited units of Table 4-1 shall employ spacing areas no less than those defined in Table 4-3. All safe units shall employ a minimum spacing between units of twelve inches.

  (3) Each safe unit shall be centered in its respective spacing area.

(b) When the above criteria for the surface density model cannot be met, the spacing may be established by the solid angle method of TID-7016 (Rev. 2) providing that the applicable criteria on subcriticality of the primary unit and subtended solid angle of interacting units are met.

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

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

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

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

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

(h) All computer analysis methods shall be validated in accordance with the criteria of Section 4.2.3.2 and Regulatory Guide 3.4, Revision 4, 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 shall be less than or equal to 0.95 including applicable biases and calculational uncertainties.

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

4.2.3 Technical Data and Validation of Calculational Methods

4.2.3.1 Technical Data

Safe unit limits which meet the subcriticality criteria for spacing by the surface density method are listed in Table 4-1. Minimum spacing criteria are as listed in Table 4-3.

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

Table 4-2 provides safe limits for aqueous solutions with enrichments up to 5 w/o U-235. The uranyl fluoride data may be used for UO_4.

A 35 Kg mass limit may be employed for homogeneous or heterogeneous UO2 in a covered, 5 gallon, or less, metal container. Heterogeneous UO2 shall include hard, clean scrap, i.e., broken pellets and chips; hard contaminated scrap, i.e., broken pellets and chips admixed with possible moderating media, shall be limited to the SIU mass values listed in Table 4-1. These containers shall be separated by a minimum of 12 inches, edge to edge, in a planar array.

4.2.3.2 Other Criteria

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

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

(c) A 35 Kg mass limit may be employed for homogeneous or heterogeneous UO2 in a covered, 5 gallon, or less, metal container. Heterogeneous UO2 shall include hard, clean scrap, i.e., broken pellets and chips; hard contaminated scrap, i.e., broken pellets and chips admixed with possible moderating media, shall be limited to the SIU mass values listed in Table 4-1. These containers shall be separated by a minimum of 12 inches, edge to edge, in a planar array.

4.2.4 Special Controls

The following technical criteria shall be employed.

a) Process areas containing fissile materials will not have fire sprinkler systems unless the fissile material is contained in approved shipping containers. Water hoses shall not be used to fight fires in moderator control areas.

b) The hygrometers on the plant air to the Receivers in the Oxide Building and to the micronizers and blenders in Building 254 will be set to alarm at a dewpoint no higher than 0 °C and checked on a 6 month interval. The hygrometer on the cooler hopper at the exit of the screw cooler in the oxide building will be set to alarm at a dewpoint no higher than 15 °C and checked on a 6 month period. Upon alarm, automatic or manual action stops the process. The source of alarm must be investigated and the problem corrected before the process can be continued.

c) The R-2 steam line will have two (redundant) fail-safe shut-off valves, each activated by two independent high and low temperature alarm setpoints on the R-2 reactor. The operability of this system will be ascertained at least once every 6 months.

d) The moisture content of the UO2 powder transferred into the bulk storage hoppers and the recycle storage hoppers will be verified as being ≤ 1 w/o. The instruments used for measuring moisture in UO2 shall be calibrated on a 6 month interval. Loading and unloading of hoppers shall be done with hoods that prevent water ingress.

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

f) The two vertical dissolver vessels in the Recycle/Recovery Area (240-2) shall have a barrier to insure that no significant moderating material can be brought within 1 foot of the cylindrical tank surface.

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

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

i) The number of 5 gallon or less containers allowed on the second and third floors of Building 254 shall be limited as follows: lubricant and/or poreformer, 12 on each floor; UO2 powder, 24 spaced on 2 foot centers on each floor. Additionally, 5 gallon or less containers of water, cleaning solutions or powder moderators (exclusive of lubricant and poreformer) in storage or use will be limited to two on the second floor and two on the third floor when the poreformer or lubricant mixing operations have material in process.

j) UO2 powder charges added to each poreformer mixer in Building 254 shall not exceed 4.4 Kg U-235.

k) Fissile aqueous solution transfers from safe to unsafe geometry vessels shall have at least two independent methods for control of the fissile content of the solution prior to release of the solution to the unsafe geometry vessel; solution transfers shall be limited such that the unsafe vessels never contain more than a fraction of the calculated critical mass. Physical barriers shall exist to prevent the inadvertent transfer of fissile aqueous solutions to unsafe geometry vessels.

l) All process systems shall be designed to minimize the likelihood for accumulation of fissile material within the system. In addition, process procedures shall have provisions for verifying that fissile material has not accumulated within the system, especially in those systems employing unsafe geometry containers.

m) The isotopic content of all incoming containers shall be verified prior to release of the container contents to systems designed for enrichments less than or equal to 5 w/o U-235.

TABLE 4-1

SAFE UNIT LIMITS

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

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

TABLE 4-2

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

	UO_2F_2	$UO_2(NO_3)_2$
Mass (Kg U-235)*	0.82	1.77
Cylinder Diameter (in.)*	10.0	16.4
Slab Thickness (in.)*	4.42	8.33
Volume (liters)*	26.9	105.5
Concentration (g U-235/L)		
Critical Limit	273	298
Subcritical Limit**	261.0	283.0

* With safety margins
** ANSI/ANS-8.1-1983

TABLE 4-3

MINIMUM SPACING AREAS[1] FOR HOMOGENEOUS AND HETEROGENEOUS MASS AND GEOMETRIC LIMITS

	Spacing Area (ft^2)
Mass	3.5
Volume	9.0
Cylinder [2]	5.0

(1) Subject to a minimum edge-to-edge unit separation of 12 inches.

(2) Per foot of cylinder height.

CHAPTER 5 ENVIRONMENTAL PROTECTION

5.1 Effluent Control Systems

5.1.1 ALARA Commitment

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

5.1.2 Air and Gaseous Effluents

Exhaust air effluents from process areas and process equipment shall be sampled continuously during operations. These stack samples shall be changed at least weekly, except that new operations shall be sampled more frequently until effective control is assured. All samples shall be counted after suitable delay for decay of radon daughters, and the results evaluated. The lower limit of detection shall be less than 5% of 10 CFR 20, Appendix B, Table II, limits. If at any time it is determined that the counting system fails to meet the lower limit of detection requirement, an investigation will be conducted and the appropriate corrective actions implemented.

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

5.1.3 Liquid Effluents

Levels of contamination in liquid effluents shall be measured by representative grab sampling of batch discards, by proportional sampling of continuous discharges, or both. Samples shall be collected at or prior to the point of discharge from the waste handling system. Samples shall be analyzed for gross alpha and gross beta activity. The lower limit of detection shall be less than 5% of 10 CFR 20, Appendix B, Table II, limits. If at any time it is determined that the counting system fails to meet the lower limit of detection requirement, an investigation will be conducted and the appropriate corrective actions implemented.

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

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

CHAPTER 7 DECOMMISSIONING PLAN

Combustion Engineering reaffirms that, upon terminating activities involving materials authorized under license SNM-33, affected facilities will be decommissioned in a manner that will protect the health and safety of the public. Combustion Engineering's Decommissioning Plan dated January 12, 1979, and financial assurances in the letter dated July 19, 1990 should be considered a part of this renewal application.

CHAPTER 8 RADIOLOGICAL CONTINGENCY PLAN

Combustion Engineering, Inc. shall maintain and execute the response measures of the Radiological Contingency Plan for the Hematite facility submitted to the NRC by letters dated December 28, 1987 and August 23, 1990 and incorporated as part of this license application. Combustion Engineering, Inc. shall also maintain implementing procedures as necessary to implement the Plan. No change shall be made in this Plan that would decrease its response effectiveness without prior approval of the NRC as evidenced by a license amendment. Changes which do not decrease the response effectiveness of the Plan may be made without prior NRC approval; such changes to the Plan shall be reported to the NRC within 6 months after the change is made.

CHAPTER 10 FACILITY DESCRIPTION

10.1 Plant Layout

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

In the Oxide Building, UF_6 is converted into UO_2 granules that subsequently follow one of two paths. UO_2 granules are transferred to bulk storage hoppers. The hoppers are then transferred to storage or for use in one of the two pelletizing buildings. UO_2 may also be loaded into approved shipping containers for shipment off site.

Building 254 has two parallel pellet lines as the primary pelletizing building. Building 255 contains one pelletizing line used as a backup and for special pellet runs.

In the pelletizing buildings, the granules flow by gravity from the hopper into a vibratory feeder which feeds the granules to the mill (micronizer). The resulting powder is vacuum transferred into a blender. The blended powder is vacuum transferred into a dry powder preparation process. A poreformer is added and blended into a batch of blended powder. Each batch of powder is pressed into slugs and then granulated. Die lubricant is mixed into the granulated powder. The blended granulated powder is gravity fed into a feed hopper on a rotary press and pressed into pellets. "Green" pellets are processed through a dewaxing furnace to remove the additives and then passed through a sintering furnace where they densify and achieve the desired characteristics. The sintered pellets are ground to the final specified diameter. Pellets may either be packaged for shipment off site or dried for fuel rod loading on site.

Support operations for the conversion and pelletizing processes include material recycle, scrap recovery, cylinder heel recovery, quality control laboratory, maintenance, waste consolidation and disposal, and effluent processing. Building 256 is the site warehouse for shipping pellets and powder and for receiving site supplies. Building 253 contains various site utilities, storage and offices. Building 240 contains laboratory and maintenance areas, a recycle recovery area, a waste incineration area and offices.

10.2 Utilities and Support Systems

10.2.1 Electric Power

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

A diesel powered emergency generator provides backup emergency power to maintain critical loads such as emergency air, water, steam, instrumentation, alarms, etc. The unit is located in the emergency utilities Building 115 to the east of Building 110. The generator is switched from normal power to generator (emergency) power by an "Onan" automatic transfer switch. Startup and transfer of the unit takes about 5 seconds. The generator is normally startup tested on a weekly basis, except during holidays or periods when the facility is shut down for an extended maintenance outage. The generator ratings and the emergency loads are listed in Table 10-1.

10.2.2 Compressed Air

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

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

10.2.3 Water

Water used on the C-E Hematite site is supplied by a well located north of Building 253 within the fenced manufacturing area. One the average day, 36,000 gallons are withdrawn from this well. The electric power supply for the well pump is automatically transferred to the emergency generator upon a power outage.

Well water is stored in an elevated 200,000 gallon tank and distributed as needed within the plant, primarily for process water. A circulating water cooling system, including a forced convection evaporative cooling tower, provides equipment cooling. A small amount of equipment utilizes once through cooling.

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

10.2.4 Sanitary System

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

The routing of the sanitary system drains is shown in Figure 10-2. The system includes a sewage treatment plant in which sanitary sewer effluents are treated with dry chlorine tablets, discharged to a chlorine contact tank and finally treated with sodium sulfite for dechlorination before discharge into the environment. Design capacity of the treatment plant is 8,000 gallons per day. The sanitary effluent enters the site creek immediately below the site pond. It is sampled and analyzed for gross alpha and beta activity. Discharge from the treatment plant is authorized under a National Pollutant Discharge Elimination System Permit To Discharge issued by the Missouri Department of Natural Resources.

10.2.5 Storm Water System

Water from roof and ground surface drains flows to the site pond above the dam via the storm water system. This system also receives condensed steam from the UF_6 vaporizer steam jackets, cooling water from heat exchanges in the recycle/recovery process and laboratory sink water from cleaning glassware.

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

10.2.6 Chemical Storage

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

Ammonia - approximately 620,000 pounds used per year as a reducing gas in the production of UO_2 powder, pellets, and in preparation of material for recycle. Typical quantity stored on site: <10,000 gallons.

Liquid Nitrogen - approximately 10,000 liters per year used with ammonia to establish a reducing atmosphere in the conversion process and the pellet furnaces. Typical quantity stored on site: <8,000 gallons.

Potassium Hydroxide - approximately 3,500 pounds used per year. Mixed with process water and used as wet scrubber liquor to remove hydrofluoric acid from the recycle pyrohydrolysis process effluent. Typical quantity stored on site: <4,000 pounds.

Hydrochloric Acid - approximately 850 pounds used per year in cleaning heat exchanger tubes in the steam boiler. Typical quantity stored on site: <1,000 pounds.

Nitric Acid - approximately 9,850 pounds used per year to dissolve the U_3O_8 wet recovery process feed material. Typical quantity stored on site: <10,000 pounds.

Hydrogen Peroxide - approximately 20,100 pounds per year used to adjust pH in the wet recovery process. Typical quantity stored on site: <10,000 pounds.

Trichlorethane - approximately 9,500 pounds per year used in preparing UO_2 powder for pelletizing. Typical quantity stored on site: <3,000 pounds.

Figure 10-3 shows where these chemicals are typically stored.

10.3 Ventilation Systems

The Oxide Building is heated by forced hot air supplied by a natural gas-fired heater located on the roof of Building 255. Buildings 255 and 240 are heated by steam supplied by the site boiler located in the south end of Building 253. This boiler is natural gas-fired from a non-interruptable supply. Buildings 253 and 254 are heated by forced hot air supplied by natural gas fired heaters located on their roofs. Only the offices, Laboratory, and Maintenance Shop are air conditioned.

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

The air cleaning equipment listed in Section 3.2.2 of this license is used in the following areas:

a. Cyclone Collectors
   - R3 reactor off-gas ahead of back-up filter.
   - Discharge of three furnace scrubbers in recycle area.

b. High Efficiency Particulate Air Filters
   - All exhaust ventilation with the exception of moist discharge from scrubbers in recycle/recovery, incineration, and UF_6 scrubber in conversion.

c. Wet Scrubbers
   - Recycle/Recovery furnaces, incinerators, NOx scrubber for wet recovery, and UO_4 scrubber for precipitation.

d. Dry scrubbers
   - Offgas from conversion (HF removal)

e. Fabric Filters
   - Pre-filters on transport operations in conversion and pellet plant, and for vacuum power removal at presses and area cleanup.

f. Special Filters
   - Sintered metal filters for high temperature particulate removal from conversion offgas.

All exhaust stacks are continuously monitored when in operation. The locations and flow rates for the exhaust stacks are shown on Figure 10-4.

10.4 Radioactive Waste Handling

10.4.1 Liquid Wastes

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

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

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

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

10.4.2 Solid Wastes

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

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

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

10.5 Fire Protection

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

Facilities are designed, constructed, and operated consistent with requirements of all applicable fire safety codes. A copy of the Certificate of Insurability from American Nuclear Insurers is provided in Figure 10-5.

10.5.1 Fire Protection Equipment

A 200,000 gallon elevated tank provides the water supply for fire hydrants and sprinklers. An electric driven fire pump in the Emergency Utilities Building 115 feeds fire hose connections. The fire pump is powered by the emergency diesel generator in the event of loss of electrical power. There are fire hose connections at each of the four corners of the plant, such that the maximum hose length to reach all areas is less than 500 feet.

There are three sprinkler systems: in the Building 255-3 Store Room, the Building 240-1 Laundry and the Building 256 Warehouse. The sprinkler systems are fed from the fire pump main, with the exception of that in the Laundry which is gravity fed from the elevated water tank. Moderator controlled areas do not have sprinkler systems, nor are fire hoses allowed in those areas. Portable fire extinguishers (carbon dioxide, dry chemical, and halon types) are located throughout the facility. Extinguishers are inspected on a periodic basis as specified by the National Fire Protection Association Code.

A centralized fire alarm system located in Building 253 provides status indication of smoke detectors, sprinkler systems, the fire pump and fire alarm pull stations. A remote fire alarm panel is provided in the Guard Station, Building 110. Smoke detectors are provided in the ventilation systems, upstream of HEPA filters. Fire alarm pull boxes are located throughout the plant.

TABLE 10-1
ELECTRICAL POWER SOURCES

Normal Power Supply Transformers

Transformer Location	Input Voltage	Output Voltage	VA Rating
East of Bldg. 110	12.5 KV	480 v	2000 KVA
East of Bldg. 110	12.5 KV	480 v	1500 KVA
Bldg. 230	12.5 KV	480 v	1500 KVA
Bldg. 240	12.5 KV	230 v	500 KVA
Bldg. 255	480 v	208 v	500 KVA
Oxide Plant	12.5 KV	480 v	500 KVA

Emergency Generator Loads

Generator Location	Rating	Primary Loads
Emergency Utilities Building	3-phase 480 volts 600 KW Startup and transfer 5 seconds	1. Instrumentation 2. Alarms 3. Well Pump 4. Nuclear Alarms 5. Burner Blower - Boiler 6. Feed Water Pump - Boiler 7. Air Compressor 8. Emergency Lighting - Buildings 230, 253, 254 and Oxide 9. Telephones 10. Cooling Water Pump 11. Cooling Tower Fan 12. Process Ventilation Blowers

TABLE 10-2
COMPRESSED AIR SUPPLY

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

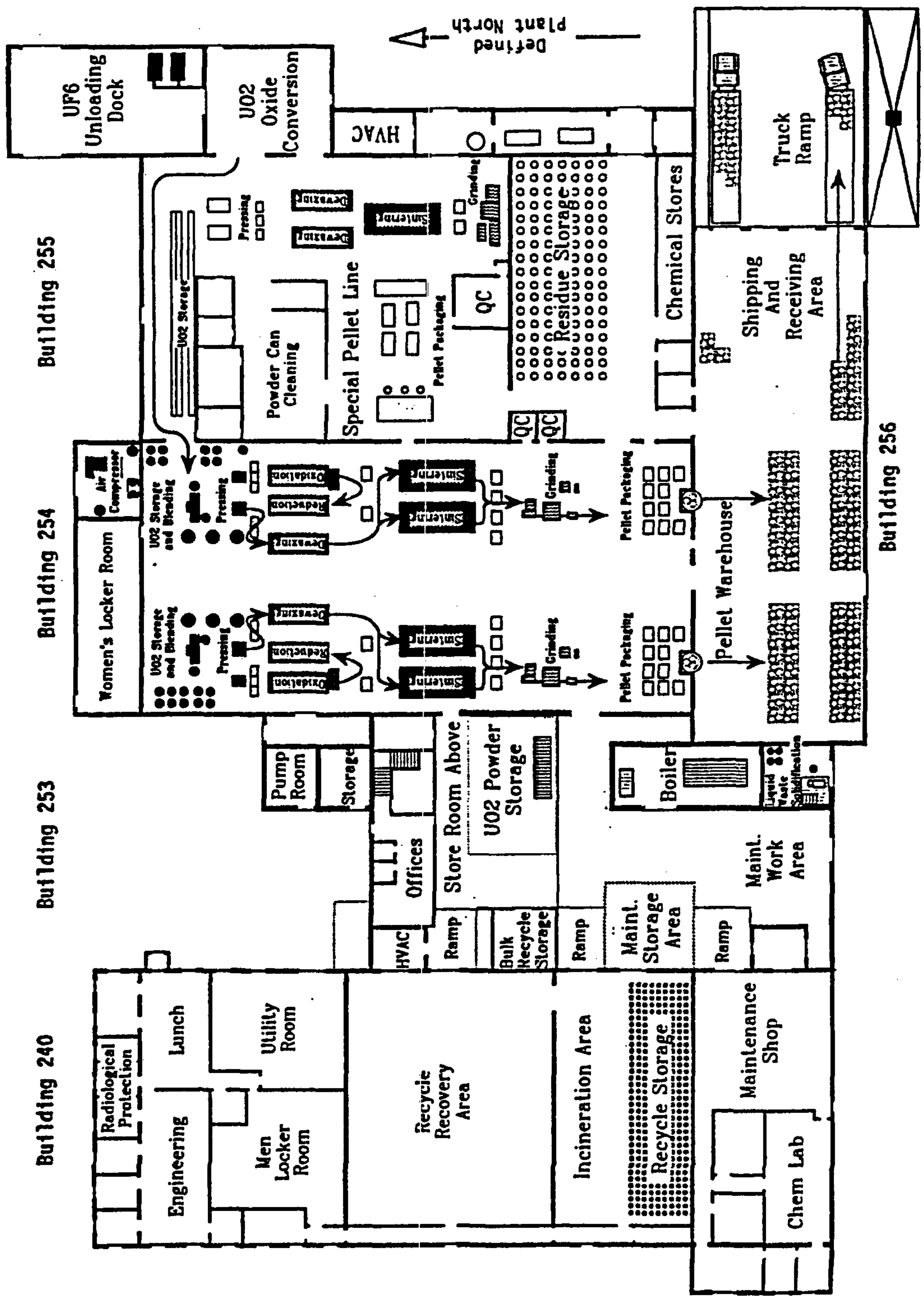


FIGURE 10-1 BUILDING AND EQUIPMENT LAYOUT

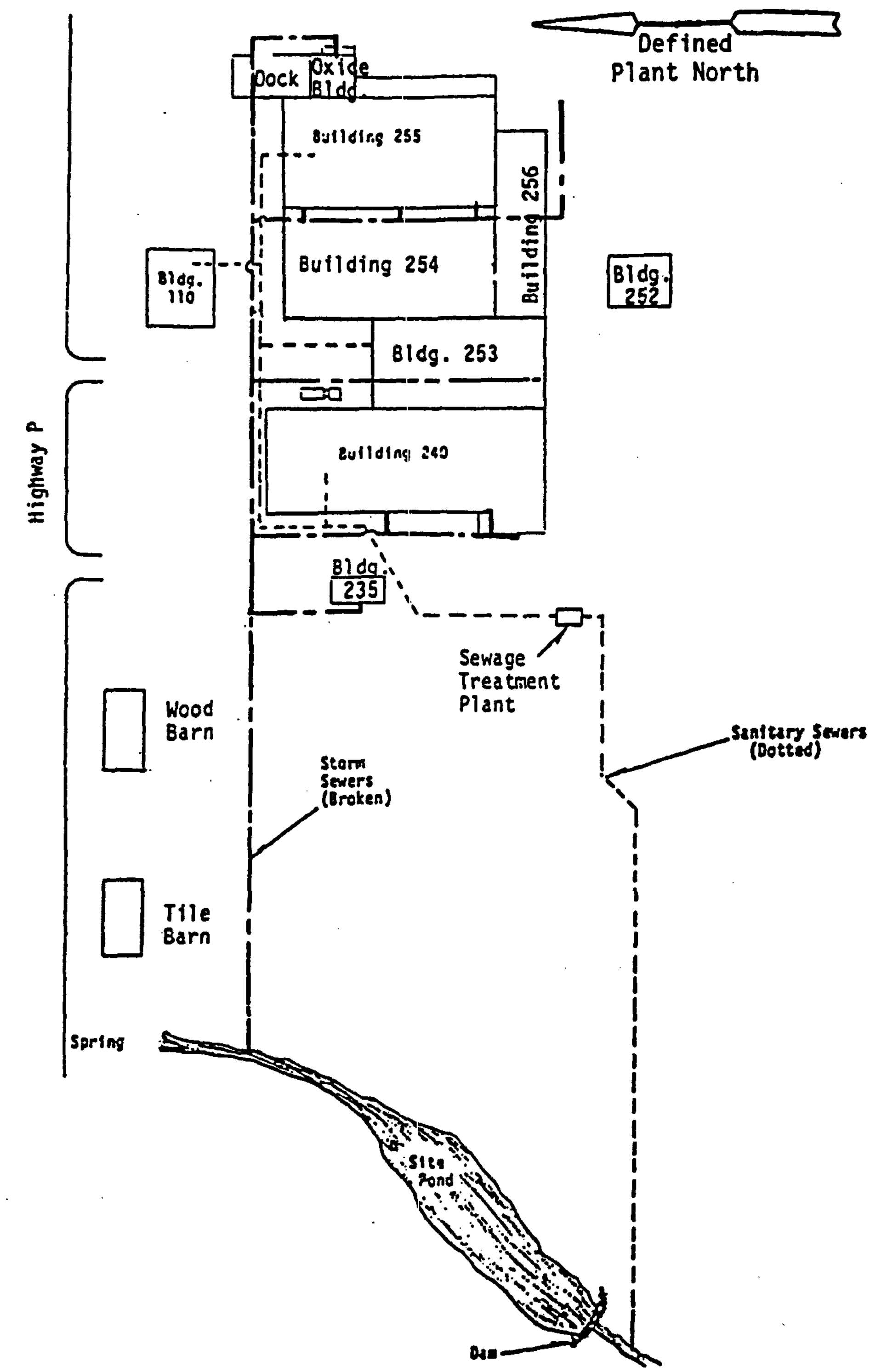
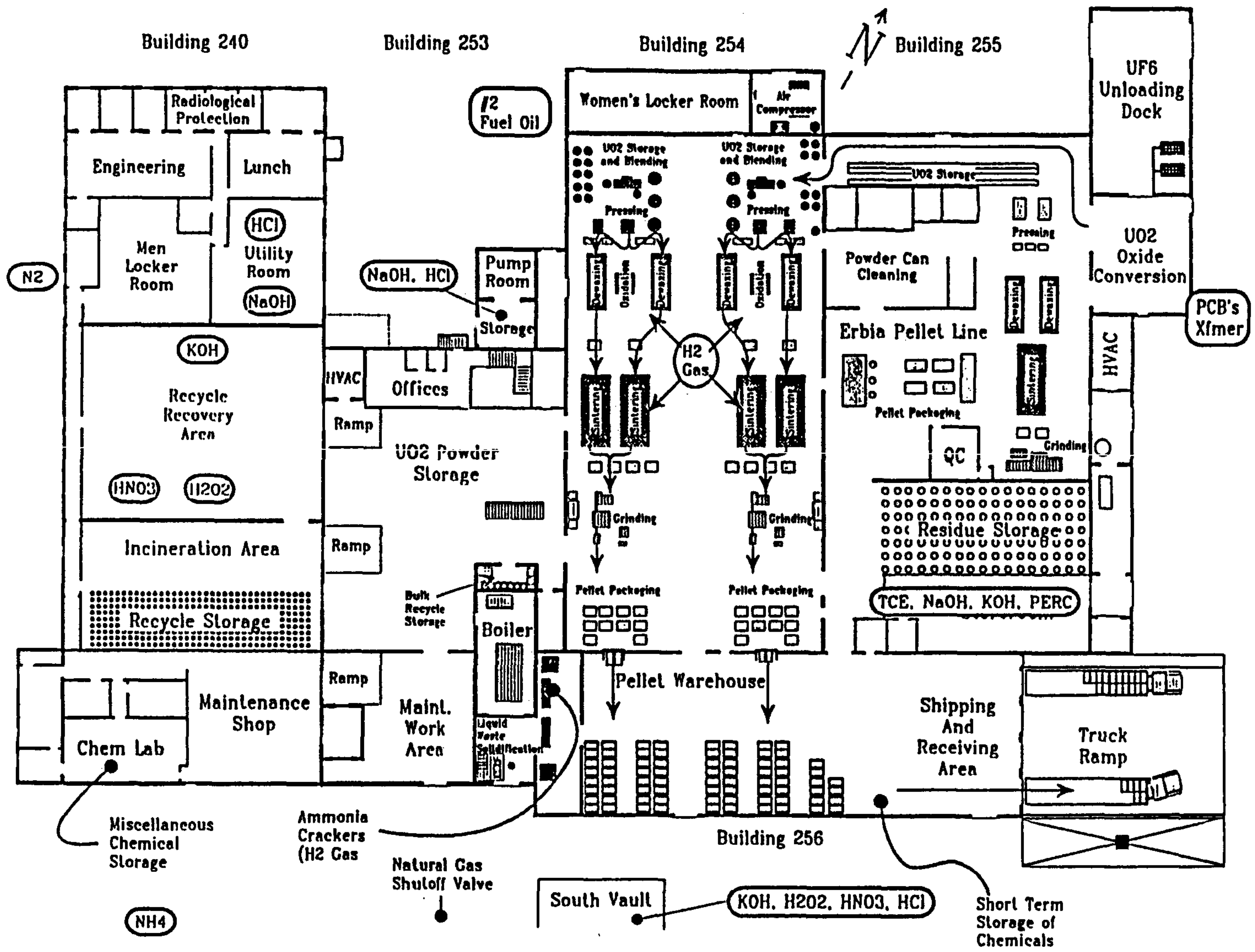


FIGURE 10-2 SANITARY AND INDUSTRIAL WASTE LINE FLOWS

FIGURE 10-3 CHEMICAL AND OTHER HAZARDOUS MATERIAL STORAGE LOCATIONS



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

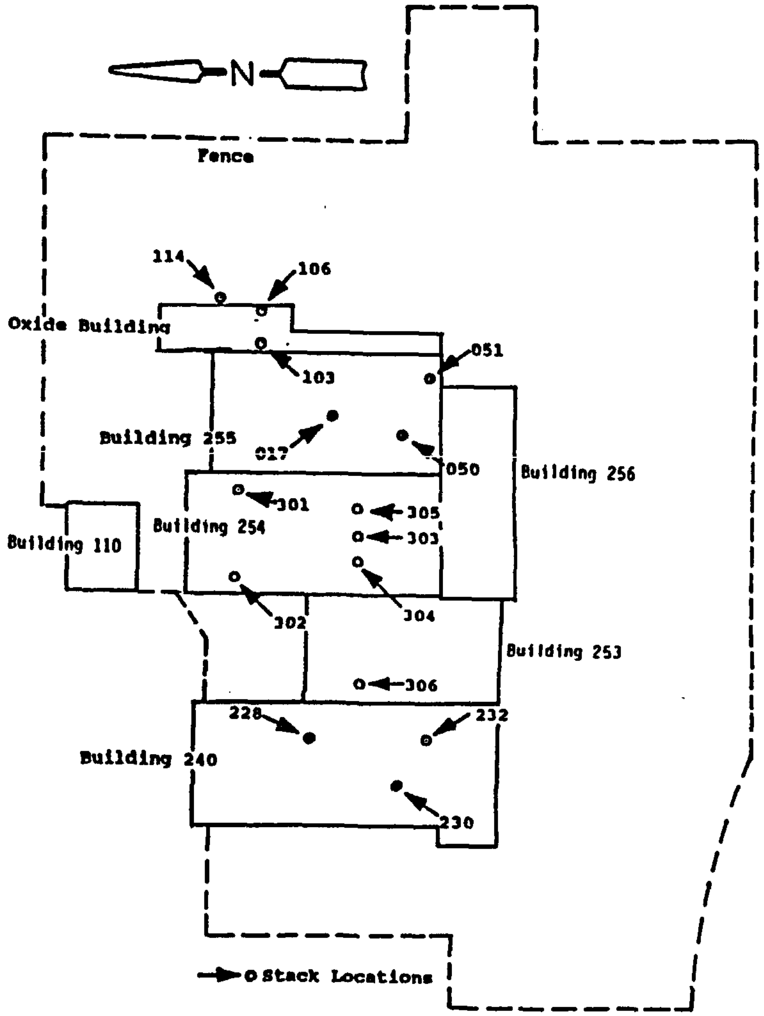


FIGURE 10-4 EXHAUST STACK LOCATIONS AND FLOW RATES

AMERICAN NUCLEAR INSURERS

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

MUTUAL ATOMIC ENERGY LIABILITY UNDERWRITERS

Suite 3720
One East Wacker Drive
Chicago, Illinois 60601

CERTIFICATE OF INSURANCE

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

Name of Insured COMBUSTION ENGINEERING, INC.

Mailing Address 900 Long Ridge Road, Stamford, Connecticut

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

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

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

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

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

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

Effective date of the Certificate: July 1, 1989

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

FIGURE 10-5 CERTIFICATE OF INSURABILITY

11.3 Education and Experience of Key Personnel

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

1. J. A. Rode - Plant Manager
2. R. J. Klotz - Nuclear Criticality Specialist (located in Windsor, CT)
3. H. E. Eskridge - Manager, Nuclear Licensing, Safety, and Accountability
4. A. J. Noack - Superintendent, Production
5. R. W. Griscom - Manager, Engineering

7. E. W. Criddle - Supervisor, Health Physics

JAMES A. RODE - PLANT MANAGER, HEMATITE

Ex 6

EDUCATION:

B.S., Chemical Engineering, University of Texas/[redacted]

EXPERIENCE:

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

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

GULF UNITED NUCLEAR FUELS CORPORATION 1968 to 1974
Technical Consultant

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

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

Responsible for design, construction and startup of the first large scale fluidized-bed process for the production of UO2 from UF6 and of companion facilities for converting oxide to pellets.

Responsible as Technical Director for Chemicals Operations for process engineering supervision and development activities including design, construction, and operations of a pilot plant for preparation of UO2 via the reaction of UF6 and steam and for development, design, construction and startup of a fluid-bed vapor phase coating system.

Assistant Technical Director 1962 to 1964

Responsible for process and equipment design in the Rhode Island Scrap Recovery Facility, development work on process for producing pyrolytic carbon coated UO2, and for continuing development work in Naval Fuel Program.

Project Leader 1961 to 1962

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

JAMES A. RODE (continued)

MALLINCKRODT CHEMICAL WORKS
Group Leader and Production Superintendent 1958 to 1961

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

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

Process Engineer and Research Chemist 1953 to 1958

Participated in preparation of proposals for production of yttrium metal and conversion of 5000 tons per year of UF6. Responsible for operation of the first ADU pilot plant and startup of the Hematite Oxide Plant.

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ROBERT J. KLOTZ - NUCLEAR CRITICALITY SPECIALIST

EDUCATION

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

EXPERIENCE

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

Senior Consulting Physicist — 1977 to Present

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

Section Manager, Radiation and Criticality Physics — 1965 to 1977

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

GENERAL NUCLEAR ENGINEERING CORPORATION
Physicist — 1959 to 1965

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

CONVAIR DIVISION OF GENERAL DYNAMICS
Physicist — 1954 to 1959

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

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

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EDUCATION:

B.S., Physics, North Carolina State University, [redacted]
M.S., Physics, North Carolina State University, 1963

EXPERIENCE:

COMBUSTION ENGINEERING, INC.

Manager, Nuclear Licensing, Safety and Accountability - Hematite	1989 to Present
Supervisor, Nuclear Licensing, Safety and Accountability - Hematite	1974 to 1989

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

GENERAL ELECTRIC COMPANY — 1972 to 1974
Nuclear Safety Engineer

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

SALISBURY METAL PRODUCTS COMPANY — 1971 TO 1972
Co-Manager

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

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

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

HAROLD E. ESKRIDGE (continued)

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

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

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

Technical, policy, and procedural consultation in all aspects of health physics, environmental surveillance and radiological health. Functioned as administrator of Radioactive Materials Licensing and Regulation. Served as Team Chief of State

Radiological Emergency Team and established and equipped a laboratory for radiological and chemical analysis of environmental samples.

U.S. AIR FORCE 1954 to 1957
Nuclear Specialist

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

ARLON J. NOACK - PRODUCTION SUPERINTENDENT, HEMATITE

<u>EDUCATION</u>:

Hillsboro High School, [redacted] Graduate

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<u>EXPERIENCE</u>:

COMBUSTION ENGINEERING, INC.
<u>Production Superintendent - Hematite</u> 1981 to Present

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

<u>Maintenance Supervisor - Hematite</u> 1980 to 1981

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

<u>Production Supervisor - Hematite</u> 1974 to 1980

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

GULF UNITED NUCLEAR FUELS CORPORATION
<u>Production Supervisor - Hematite</u> 1970 to 1974

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

<u>Engineering Technician - Hematite</u> 1969 to 1970

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

UNITED NUCLEAR CORPORATION 1966 to 1969
<u>Process Development Technician - Hematite</u>

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

LUDLOW SAYLOR WIRE CLOTH COMPANY 1963 to 1966
<u>Production Operator - St. Louis</u>

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

ARLON J. NOACK (continued)

HOWARD INDUSTRIES COMPANY 1962 to 1963
Junior Draftsman - Festus

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

Ex 6

ROBERT W. GRISCOM - MANAGER, ENGINEERING, HEMATITE

<u>EDUCATION</u>:

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

<u>EXPERIENCE</u>:

COMBUSTION ENGINEERING, INC.	1981 to Present
<u>Manager, Engineering - Hematite</u>	1989 to Present
<u>Engineering Supervisor - Hematite</u>	1981 to 1989

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

NATIONAL STEEL ENGINEERS & ASSOCIATES	1977 to 1981
<u>Project Manager - St. Louis</u>	

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

ROCKWELL INTERNATIONAL	1974 to 1977
<u>Senior Test Engineer - St. Louis</u>	

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

MONSANTO COMPANY	1969 to 1974
<u>Process Engineer - St. Louis</u>	

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

Ex 6

ENOS W. CRIDDLE - SUPERVISOR, HEALTH PHYSICS

Cape Girardeau Central High School, Graduated [redacted]
Naval Nuclear Power School, 1982
Naval Nuclear Power Prototype Training, 1983
Naval Nuclear Engineering Laboratory Technician, 1983
Naval Damage Control School, 1984
Naval Fire Fighting Training, 1985

PROFESSIONAL EXPERIENCE:

<u>ABB Combustion Engineering Nuclear Power, 1988 to Present</u>

<u>Health Physics Supervisor, 1990 to Present</u>

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

<u>Health Physics Technician, 1988 to 1990</u>

Responsible for radiological and industrial safety at Nuclear Fuel manufacturing - Hematite. Duties include instrument calibration, environmental sampling, documenting employee exposures, maintaining health physics documents, and performing routine radiological and industrial safety monitoring.

<u>U.S. Navy Engineering Laboratory Technician, 1981 to 1987</u>

Stationed on board USS Lafayette SSBN 616 (G) responsible for radiological safety throughout the ship. Qualified supervisor for administration and control of radiological materials and records. Responsible for instrument and gauge calibration program, chemical inventory and storage, and water chemistry controls for reactor plant and steam plant.

11.4 Operating Procedures

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

11.5 Training

The training program for employees is described in Section 2.5.

11.6 Changes in Procedures, Facilities and Equipment

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

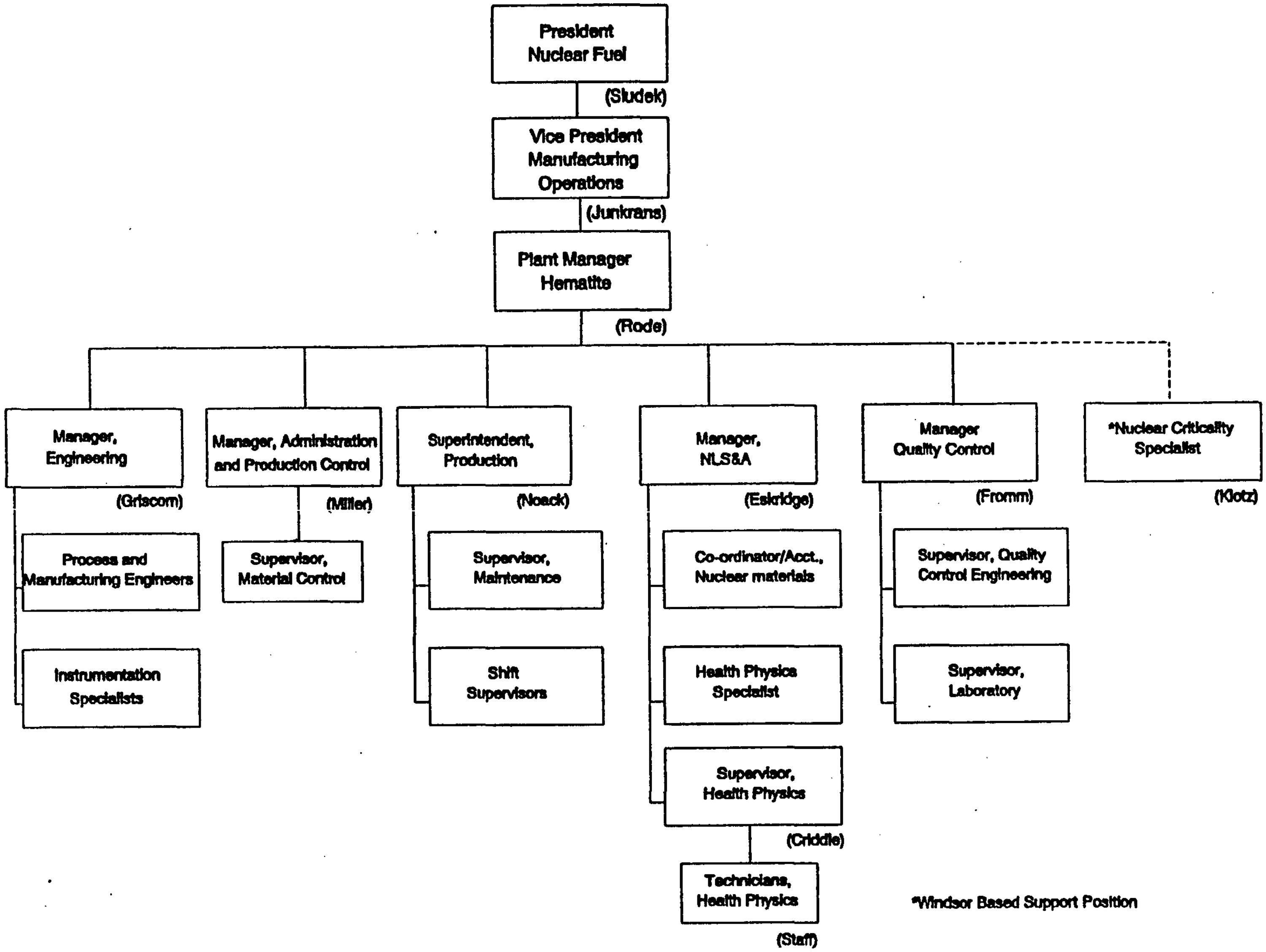

FIGURE 11-1 HEMATITE PLANT ORGANIZATION CHART

CHAPTER 12 RADIATION PROTECTION

12.1 Program

The ALARA policy is described in Section 3.1.1. A manual containing procedures necessary to implement the radiation safety program described in Part I of this renewal application is maintained by NLS&A. Following are some of the types of procedures which are used:

Fixed Workstation Air Sample Collection and Logging
Exhaust Stack Sampling
Lapel Air Sampling
Hydrogen Fluoride Sampling of the Oxide Scrubber Exhaust
Annual Ventilation System Stack Velocity Measurements
Ventilation System Filter Efficiency Measurement
Performing Smear Surveys
Performing Trailer Surveys
Surveying Items for Release from the Plant
Survey of Incoming UF_6 Shipments
Quarterly Beta/Gamma Survey
Calculating Employee Weekly Cumulative Exposures
Employee Bioassay Sampling
Instrument Calibration
Alarm Testing
Environmental Sampling: Water, Soil, Vegetation and Air

All routine operations involving SNM handling are covered by an Operation Sheet (O.S.) and/or by a Special Evaluation Traveller (S.E.T.). A separate O.S. covers plant-wide radiation safety procedures, while procedures specific to a certain operation are covered in the O.S. for that operation. O.S.s and S.E.T.s prepared for operations/processes involving radiological safety are reviewed for consistency with the ALARA policy identified in Section 3.1.1.

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

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

12.2 Posting and Labeling

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

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

12.3 External Radiation - Personnel Monitoring

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

12.4 Radiation Surveys

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

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

12.5 Reports and Records

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

Retention of records is described in Section 2.9.

12.6 Instruments

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

12.7 Protective Clothing

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

12.8 Administrative Control Levels, Including Effluent Control

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

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

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

12.9 Respiratory Protection

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

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

12.10 Occupational Exposure Analysis

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

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

12.10.1 External Radiation Exposures

The exposure to radiation from external sources is measured using film badges. The film badges are changed monthly. Results of monitoring for 1989, 1990, and 1991 were as follows:

	Percent of Personnel in Range		
Annual Dose Ranges, gamma (REM)	1989	1990	1991
No measurable exposure	22	33	37
Less than 0.100	43	38	42
0.100 - 0.250	26	19	21
0.250 - 0.500	8	9	10
0.500 - 0.750	0	1	1
0.750 - 1.000	1	0	0
Number of employees monitored	106	111	111

12.10.2 Internal Radiation Exposures

Air concentration levels are measured using Breathing Zone (BZ) monitors. Fixed Work Station Air samples are used to calculate air concentration levels when BZ sampling data is not available. The quarterly average air concentrations for reporting years 1989, 1990 and 1991 were as follows:

Quarterly Exposure Range MPC hrs/% of Limit	Percent of Operators in Range											
	1989				1990				1991			
	1st	2nd	3rd	4th	1st	2nd	3rd	4th	1st	2nd	3rd	4th
Oxide Plant Operators												
0 - 52 / 0% - 10%	0	0	0	0	12	0	0	0	0	0	0	0
52 - 130 / 10% - 25%	22	22	22	75	44	89	44	22	33	67	33	90
130 - 260 / 25% - 50%	78	78	67	25	44	11	56	78	67	33	67	10
260 - 520 / 50% - 100%	0	0	0	0	0	0	0	0	0	0	0	0
Pellet Plant Operators												
0 - 52 / 0% - 10%	0	8	8	0	0	25	19	6	7	0	0	5
52 - 130 / 10% - 25%	77	77	92	92	100	75	69	47	40	69	59	84
130 - 260 / 25% - 50%	23	15	0	8	0	0	12	47	53	31	41	11
260 - 520 / 50% - 100%	0	0	0	0	0	0	0	0	0	0	0	0
Recycle Plant Operators												
0 - 52 / 0% - 10%	0	17	33	17	33	14	17	0	0	17	0	0
52 - 130 / 10% - 25%	67	83	67	83	67	86	66	50	80	50	78	56
130 - 260 / 25% - 50%	33	0	0	0	0	0	17	50	20	33	22	44
260 - 520 / 50% - 100%	0	0	0	0	0	0	0	0	0	0	0	0
Material Control Operators												
0 - 52 / 0% - 10%	100	100	100	100	100	100	67	0	25	25	25	50
52 - 130 / 10% - 25%	0	0	0	0	0	0	33	100	75	75	75	50
130 - 260 / 25% - 50%	0	0	0	0	0	0	0	0	0	0	0	0
260 - 520 / 50% - 100%	0	0	0	0	0	0	0	0	0	0	0	0
Utility Operators												
0 - 52 / 0% - 10%	0	10	30	11	12	17	38	0	13	0	0	10
52 - 130 / 10% - 25%	40	70	70	78	88	83	62	92	60	80	50	70
130 - 260 / 25% - 50%	60	20	0	11	0	0	0	8	27	20	50	20
260 - 520 / 50% - 100%	0	0	0	0	0	0	0	0	0	0	0	0
Maintenance												
0 - 52 / 0% - 10%	0	25	25	12	30	25	10	0	11	0	10	0
52 - 130 / 10% - 25%	100	75	75	88	70	75	90	100	56	75	70	88
130 - 260 / 25% - 50%	0	0	0	0	0	0	0	0	33	25	20	12
260 - 520 / 50% - 100%	0	0	0	0	0	0	0	0	0	0	0	0

The maximum quarterly exposure for 1989 was 226 MPC hrs or 44% of the allowable limits set forth in 10 CFR 20.103. For 1990 the maximum quarterly exposure was 195 MPC hrs or 38% of the quarterly limit. For 1991 the maximum quarterly exposure was 230 MPC hrs or 44% of the limit.

In-vivo counting is performed by an outside contractor using gamma scintillation spectrometry. Results for 1989, 1990 and 1991 were:

Range (μg U-235)	Apr 1989	Jul 1989	Oct 1989	Jan 1990	Apr 1990	Jul 1990	Oct 1990	Jan 1991	Apr 1991	Jul 1991	Oct 1991
	Percent of Operators in Range										
Less than 50	12	13	17	6	7	17	16	17	21	19	20
50 - 100	4	7	3	2	4	5	9	7	7	6	9
100 - 125	0	0	0	0	7	0	0	0	0	0	0
125 - 240	0	0	0	0	0	0	0	0	0	0	0
Greater than 240	0	0	0	0	0	0	0	0	0	0	0

Operators are placed on partial restriction status when they have 1/2 of a maximum permissible lung burden. At the end of 1991, no operators were on restricted status.

Urinalysis is conducted for all operators on at least a monthly schedule. Results for 1989, 1990 and 1991 were as follows:

Operation	Average Concentration (μgU/liter) 1989		1990		1991	
	Average of Conc.	Number of Operators	Average of Conc.	Number of Operators	Average of Conc.	Number of Operators
Maintenance	0.7	10	0.6	12	0.7	10
Material Control	0.6	5	0.6	7	0.6	4
Oxide Plant	0.6	9	1.0	9	1.3	10
Recycle Plant	0.7	3	0.5	7	0.8	9
Pellet Plant	0.6	17	0.5	15	0.5	18
Utility	0.5	9	0.9	12	0.7	10
Total Employees		53		62		61

12.11 Measures Taken to Implement ALARA

C-E has made numerous changes to equipment and procedures to improve containment and to reduce airborne exposures. Personnel radiation exposure levels are reviewed periodically by the Health Physics Supervisor and the Manager, NLS&A. This is a continuous process of evaluation and change, and reflects the C-E operating philosophy to keep occupational radiation exposures as low as reasonably achievable.

12.12 Bioassay Program

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

12.13 Air Sampling

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

12.14 Surface Contamination

Control of surface contamination is described in Section 3.2.6. Change areas, where personnel exit from contaminated areas, are normally surveyed for alpha contamination on a daily basis during use.

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

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

The dependence of the multiplication factor on horizontal bidirectional separation of each vertical pair of trays and density of water within this spacing is examined with the KENO-IV code. Sixteen broad group cross sections are generated for each region of the model using the XSDRNPM and NITAWL codes and the 123 group library (DLC-16). Results are summarized in Figure 14-20.

Conclusions drawn from these results are as follows.

(1) A fully reflected, infinite slab having a thickness of 4.0 inches, which consists of a stack of two thin walled stainless steel trays containing 0.4 inch diameter pellets having an average UO_2 density within the tray of 5.686 g/cc and the remaining volume filled with water, has an effective multiplication factor of 0.815 $\pm$ 0.008. The subcritical margin demonstrates the conservatism of the 4 inch slab limit for these containers.

(2) For a range of spacings of up to 6 inches between the 4 inch high modules of the slab and for a range of water densities in the intra-module spaces varying between zero and full density, the effective multiplication factor is less than the value with zero spacing. Thus, the introduction of extraneous moderating materials between components of a large array of pellet trays arranged in a slab configuration will not result in an increase in the effective multiplication factor.

14.4 Fixed Poisons

Holding tanks may be poisoned with Raschig rings in accordance with Regulatory Guide 3.1, Revision 2, which states that the guidance contained in ANSI Standard ANS 8.5-1986 provides a procedure generally acceptable to the NRC staff. Raschig ring sample tubes will be provided to enable inspection for accumulation of solids and to provide samples for testing the physical and chemical properties of the rings. These inspections and tests will be conducted for the present use of the Raschig ring tanks within 13 months in accordance with the ANSI Standard.

14.5 Structural Integrity Policy and Review Program

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

Moderation control is also exercised in certain operations outside the broken line of Figure 15-1 since these operations involve the addition of moderating materials to the UO_2 and the container volume may be in excess of the safe volume. In these cases, additional administrative controls and engineered safeguards are implemented to assure that the double contingency principle is adhered to.

Prevention of moisture addition is assured by a combination of equipment design, automatic instrumentation, and administrative procedures in the form of Operation Sheets. Table 15-1 lists the measurements, allowed values, and automatic or operator actions for key steps in the process lines. By these means, the principle of double contingency is implemented, leading to the conclusion that violation of the "dry" powder criterion for UO_2 hoppers and blenders is not credible. Under normal operating conditions, UO_2 additives and moisture control are rigidly monitored to ensure product quality.

The UO_2 moisture content in the receiver is measured by withdrawing a sample and using an RF Moisture Analyzer. This device measures the dielectric changes in the UO_2 sample as a function of the moisture content. Prior to measuring each sample with the moisture analyzer, standards are measured to verify the analyzer is operating satisfactorily. If the moisture content of the sample exceeds 0.3% water by weight, the sample is retested. The retest is performed using a loss of drying method which utilizes a microwave oven and a nitrogen atmosphere. If the measured water content exceeds 0.7% by weight, the UO_2 in the receiver is not released to the 1000 kilogram hopper, but is instead removed from the receiver in safe volume containers for processing.

The 200 kilogram recycle hoppers are filled with UO_2 from safe batch containers. Each safe batch container is tested for moisture content prior to emptying into the recycle hopper using the same acceptance criteria as identified above for the receivers. When the filling of the recycle hopper is complete, the UO_2 is mixed and a sample withdrawn and measured for moisture content. If the contents of the recycle hopper passes the above criteria, the hopper is released for addition to the pellet line.

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

15.2 UF_6 to UO_2 Conversion

This system is designed to convert uranium hexafluoride to UO_2 granules by a fluidized bed "dry" conversion process. The equipment is currently qualified to handle a maximum enrichment of 5.0% U-235.

15.2.1 Receive and Store UF_6

UF_6 is received from the Department of Energy and other suppliers. The certified isotopic content of UF_6 cylinders received from the Department of Energy is used without further testing. For UF_6 cylinders received from other suppliers, an independent test is performed to verify isotopic content of the cylinders.

UF_6 is received in standard 2.5 ton, 30 inch diameter cylinders in approved shipping packages. Upon receipt, the cylinders are placed on the UF_6 cylinder storage pad which holds up to 54 cylinders. Eighteen additional cylinders may be located adjacent to the vaporizers near the cylinder scale, or in shipping packages on the Oxide Building dock. As required, a UF_6 cylinder is removed from its shipping package or storage and connected to the conversion equipment. The UF_6 storage pad is separated from the UF_6 Unloading Dock building by more than 12 feet.

Individual UF_6 cylinders filled with 5.0 wt% U-235 are a safe moderation controlled subcrit as discussed in Sections 14.3.1.1.2 and 14.3.4.2. On the storage pad they are spaced 1 foot apart in a planar array. This assures a safe configuration.

15.2.2 UF_6 Conversion Process

15.2.2.1 UF_6 Vaporizers

Description - The UF_6 vaporizers are located in the UF_6 Unloading Dock building immediately adjacent to the Oxide Building (see Figure 10-1). This area is enclosed by a metal frame building with a metal roof, metal walls, and overhead doors. Vaporization of the UF_6 in a 30 inch cylinder is done by heating the cylinder in a steam chamber. There are two chambers but only one cylinder is connected on-line at a time; when one cylinder is nearly empty, the second cylinder is placed on-line. A valving arrangement prevents the two cylinders from being interconnected.

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

During normal operation of the vaporizer, the UF_6 gas leaves the cylinder through a steam traced and insulated line into the Oxide Building. It passes through metering valves and is carried vertically along the wall to the third level of the Oxide Building and directly into the conversion equipment.

Between each vaporizer and the common piping connection, there are two control valves, HCV 5-1 and HCV 5-2. These valves are operated from one switch which electrically interlocks the operation such that only one valve can be open at a time. The switch positions are: vaporizer 1 open, vaporizer 2 open, both closed.

The same type of valving arrangement is used on the piping to the cold trap. In this case a single switch operates HCV 3-1 and HCV 3-2, which are located between the vaporizers and the common piping connection to the cold trap system.

The vaporizer dimensions are: 36" ID x 7'8" with 3" thick walls. The outside dimensions of a 30B cylinder are 30" OD x 6'4". This leaves an annulus of approximately 23.1 cubic feet. The condensate drains through a 1 1/2" pipe, arranged with a P-trap which holds the conductivity probe (always submerged). The water trap is followed by a valve before discharging into the storm drain. The piping to the vaporizer is insulated with 1 1/2" thick calcium silicate insulation.

When the UF_6 flow from a given 30 A/B cylinder is insufficient to feed the R-1 conversion reactor, the cylinder is shutdown. The residual UF_6 remaining in the cylinders exceeds that permitted for the heel removal operation preceding cylinder recertification and may exceed that desired for refilling. To reduce the residual, the nearly spent cylinders are again heated in the vaporizer and the cylinder outlet is piped to the Cold Trap System. The Cold Trap System contains an 8A cylinder which is under a vacuum and chilled with an ethylene glycol solution. The vapor pressure differential between the heated 30 A/B cylinder and cold 8A cylinder will cause a major portion of the UF_6 residual in the 30 A/B cylinder to be transferred to the 8A cylinder. The cold trapping of a given 30 inch cylinder is continued until the rate of mass transfer to the 8A cylinder decreases below a level specified in the operation sheet. Verification that the process was successful is obtained by recording the final weights of both the 30 inch and 8A cylinders. When sufficient UF_6 has been accumulated in the 8A cylinder, this cylinder is heated and the UF_6 is fed to the R-1 conversion reactor as with a 30 A/B cylinder.

<u>Safety Features</u> - In the event of a UF_6 leak into the vaporizer steam chamber, condensing steam will take SNM to the condensate line. When the conductivity cell in the drain line senses SNM, it will close the automatic shut-off valve, start the wet scrubber, and shut off the steam supply. Air, steam and UF_6 vapor from the steam chamber are ducted to the scrubber liquor in a 6 inch diameter ejector-venturi type scrubber. The separation of the condensate from the washed air is accomplished in a baffled separator, 23 inches by 9 inches by 15 inches deep. The condensate drains to a 9.75 inch internal diameter hold tank where it is recirculated to the ejector. The washed non-condensable gases are exhausted from the separator through a 6 inch duct to the atmosphere by a blower. Any overflow from the hold tank drains through a 1 inch line to a safe geometry container.

Should a leak of UF_6 occur to the atmosphere in the UF_6 Unloading Dock building, the UF_6 gas readily forms a visible cloud. A monitor on the wall alarms at the control panel, turns off the roof vent in the Unloading Dock, and the make-up air blowers in Building 255. In the event of such a leak, an emergency alarm will be sounded, the area is evacuated and the emergency procedure put into effect. Self-contained breathing apparatus and protective clothing will be worn to correct the leak. The UF_6 flow can be terminated from the control panel, at the vaporization chamber, or at the loading dock door (during evacuation). Air sampling and decontamination will be done as required to cope with suspected or actual release of airborne activity.

Valve covers are not used in handling and storage of cold UF_6 cylinders. Damage to the valve is very unlikely with the limited handling of filled containers. Even if a valve were to be broken or cracked, no significant release of UF_6 would occur. The cylinders are filled under vacuum with a substantial void space remaining. The vapor pressure of UF_6 at ambient temperature is less than the remaining vacuum so initial leakage would be inward. A crack would rapidly seal with uranyl fluoride from hydrolysis of atmospheric moisture. CO_2 to freeze the UF_6 and wooden plugs to replace a broken valve are readily available.

The cold trapping of the residual UF_6 into the 8A cylinder is normally terminated, as per the operation sheet, when the UF_6 contents of the 8A cylinder approach 60 Kg. If this limit is not observed by the operator, and the 8A cylinder content increases to 100 Kg UF_6, the system will alarm. This limit is selected to prevent hydraulic over pressure.

There are several controls which prevent the backflow of moderating materials from the conversion lines to the UF_6 cylinders. There are several valves and two differential pressure measurement orifices in the piping between the vaporizers and the R1 reactor. There are no check valves.

A differential pressure is maintained between the vaporizer and the reactor to cause UF_6 to flow into the reactor. Flow transmitter FT-5 is installed to detect loss of flow or low flow conditions. When loss of flow or a low flow condition is detected, nitrogen is introduced to purge the line between the vaporizer and the reactor. A low pressure alarm from PIT-7 provides an indication of low inlet pressure on the UF_6 line to the reactor. A low flow alarm from FT-4 indicates a low UF_6 flow as well. Protective interlocks cause the UF_6 flow control valve FCV-4 to close when any of the following conditions are detected: 1) valves are opened to supply nitrogen to the reactors; 2) low nitrogen supply pressure is detected; or 3) a low steam flow to R1 is detected.

Nuclear Safety - A leak of UF_6 into the steam chamber will not produce a criticality event because of the relatively small volume of the 3 inch thick steam chamber. For example, if it is assumed that the steam chamber is filled with UF_6 gas, it is equivalent to a layer of solid UF_6 of thickness 0.017 inches. For this purpose the density of UF_6 as a solid (5.1 g/cc) and gas (0.11 g/cc) were taken from ORO-651 (Rev. 4). Even if the system were fully reflected with water, the cylinder plus the assumed UF_6 layer on the surface of the cylinder would not pose a criticality problem. In reality the UF_6 and remaining steam would react and exit through the scrubber duct.

A second scenario concerning a leak of UF_6 into the steam chamber was analyzed. This scenario assumes the conductivity cell closed the drain line but failed to shut off the steam flow and start the wet scrubber. A mixing of steam with the UF_6 would occur, forming an aqueous solution of UO_2F_2 in the steam chamber. The steam flow is 7.9 pounds per hour. If one assumes an infinite heat sink exists at the surface of the steam chamber, it would take approximately 5.6 days to precipitate a volume of water equivalent to the volume of the steam chamber. Any escaping UF_6 would react with moisture in the air, form a visible cloud, and activate a detector which would alarm at the control panel.

An analysis was performed assuming the detector did not function properly and the steam chamber filled with UO_2F_2. The UO_2F_2 solution consists of 65.55 wt% UO_2F_2 and 35.45 wt% H_2O and has a specific gravity of 2.224 g/cc (R. Kunin, Carbide and Carbon Chemicals Co., A-3255, May 8, 1945). The mass of uranium in the steam chamber was subtracted from the uranium assumed to be left in the UF_6 cylinder, leaving 1000 kg of UF_6 in the latter container. The system was analyzed with and without a full water reflector. The K_{eff} was 0.71194 ± 0.00270 and 0.81238 ± 0.00341 for the bare and reflected cases, respectively. Even under these unlikely conditions, a criticality concern does not exist.

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

The 9.75 inch diameter hold tank on the wet scrubber is a safe geometry for UO_2F_2 (see Table 14-1).

Although the baffled separator has a total volume of 51 liters, it cannot be filled with scrubber liquor because of: (1) the drain to the hold tank, and (2) the 6 inch diameter duct on the side of the separator venting the gases to the atmosphere.

The 8A cylinder employed in the Cold Trap System is a safe cylinder for up to 115.67 Kg UF_6 net weight with a maximum enrichment of 12.5% U-235.

15.2.2.2 UF_6 to UO_2 Reactors

Description - The UF_6 to UO_2 conversion is accomplished within the high temperature environments of the three series connected reactors: R-1, R-2, and R-3. These reactors are illustrated in the schematic flow diagram of the UF_6 conversion process shown in Figure 15-2 and in the elevation sketch, Figure 15-3.

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

15.3.5 <u>Building 254 Slugging, Granulation, and Pressing</u>

Post blending process steps through pellet pressing are contained in a vertical column beginning with the blended UO_2 powder receiving hood on the third floor and ending with the rotary pellet press on the main floor (See Figure 15-10). The various steps in the process provide for addition of poreformer and press fines (plus other process line make-up materials) at the hood on the third floor, a conical screw mixer, a slugging press, a granulator, the lubricant and granulator feed hood on the second floor, the lubricant mixer, and the pellet press feed hopper. The green pellets emerging from the rotary press are loaded into sintering boats.

The mode of operation of the series of process steps illustrated in Figure 15-10 is summarized in Table 15-2 for a given batch of material starting at the air mix blender and proceeding through the rotary press. A second batch may not enter the poreformer mixer until the preceding batch has cleared the granulator and is being processed in the lubricant mixer. Thus, at any one time, up to two batches may be in process in the vertical line, one batch in the poreformer mixer and a second batch in the lubricant mixer. The valve at the outlet of the poreformer mixer is not opened until the second batch has cleared the lubricant mixer and the outlet valve closed.

The criticality analysis performed for this process demonstrated that the entire system can be full of UO_2 and not pose a criticality concern (see Section 15.3.5.4). The criticality safety, therefore, is not dependent on batch control.

It is the objective of the remainder of Section 15.3.5 to show that adequate safeguards exist in the event that postulated worst case scenarios are encountered in this moderation controlled sequence of operations. Two specific potential pathways to criticality have to be addressed since both the poreformer and lubricant mixers are not safe volumes and moderating media are normally entered via an open faced hood in the case of the poreformer mixer and a glove box in the case of the lubricant mixer. Thus, it is necessary to examine potential scenarios involving the inadvertent addition of excessive moderating media present on the operating floors.

In subsequent subsections, isolated components of the process line as well as interactive analysis between components are employed to examine the criticality characteristics and safeguards requirements for the vertical processing line.

15.3.5.1 <u>Poreformer Mixer</u>

<u>Description</u> - The poreformer mixer is a frustrum of a cone with an auger that rotates about its longitudinal axis as well as the axis of the cone. The top inside diameter of the mixer is 30 inches, the bottom inside diameter is 8 inches and height is 35.5 inches. The nominal wall thickness of the cone is 1/8 inch. The capacity of the mixer is 183.4 liters minus approximately 9.3 liters

Safety Features - The following specific procedures apply to the cylinder washing operation.

(a) No cylinder containing a heel greater than 12 Kg of UF_6 will be released for washing, as determined by weighing on the calibrated UF_6 cylinder scales.

(b) The cylinders will be washed successively with 4 gallons of the wash solution until the uranium concentration in the wash solution is <5 g U/liter. Each batch will be retained in its container until sampled.

(c) The wash solution will be sampled and on this basis consolidated into safe mass batch sizes.

(d) Operators handling wash solutions are required to wear dosimeters, chemical face shields, rubber gloves and rubber aprons. In addition, beta and gamma radiation levels on the initial wash solution for each 30 inch cylinder are measured and recorded.

Nuclear Safety - The 30 inch cylinder wash operation is safeguarded by the following administrative requirements. First, the cylinder weight is reduced by vaporization of the UF_6 directly to the conversion reactors. Second, the UF_6 is cold-trapped to an 8A cylinder until the vapor pressure of the heel is no longer sufficient to transfer further material to the 8A cylinder. Next, the wash operation is restricted to cylinders having a maximum heel weight of 12 Kg UF_6. Administrative procedures (operation sheets) require management approval for the cylinder wash operation on any cylinder having a UF_6 residual in excess of 6.8 Kg UF_6.

The maximum amount of wash solution introduced into a cylinder is limited to 4 gallons. Upon completion of each successive wash cycle, the solution is collected in a separate 5 gallon pail for each wash and all pails are spaced a minimum of 12 inches apart. Thus, the maximum uranium per pail is 0.676 times the maximum heel per cylinder (12 Kg UF_6), or 8.1 Kg U. Since the safe subcritical limits for UO_2F_2 of 5 w/o enrichment noted or inferred from Table 4-2 are 16.4 Kg U and 26.9 liters, the above process provides adequate safeguards to assure sufficient subcritical margin.

15.7.1.2 Waste Incineration

Description - The waste product incinerators are used to reduce the volume of combustible, low level uranium contaminated waste products and to incinerate combustible bulk materials having recoverable quantities of uranium.

Packaging of combustibles in bags, use of ventilated hoods to prepare charges for incineration and ash removal by vacuum collection limit exposure of personnel.

There are two dual gas burner fired incinerators. The smaller unit employs a two chamber incinerator with a flue gas to air heat exchanger to cool the gases entering the scrubber system. The second (larger) dual chamber incinerator employs the flue gas cooling/scrubber system of the smaller incinerator in parallel with a second system consisting of a water quenched flue gas cooling duct followed by a second scrubber system.

The heat exchangers and scrubbers are activated and the incinerator is pre-heated prior to introducing a charge of material into the primary combustion chamber. The flue gas to air heat exchanger discharges the heated air to a roof stack during warm weather and to the interior of Building 240 incineration area during cold weather. Scrubbed gases are discharged into stacks after the HEPA filter.

Flow rates of the incinerator exhaust stacks, No. S-230 and S-232 are given in Figure 10-4. Stack heights are 26.5 feet and 25.5 feet above grade, respectively.

The packed tower scrubbers employed here are very similar to the scrubbers used with the furnaces in the wet recovery area, except these scrubbers use ammonium hydroxide as a scrubber liquid. Thus, the same control procedures are used. The scrubber liquor is sampled weekly and analyzed for uranium concentration. The scrubber will be drained and flushed if the uranium concentration exceeds 1 g/l. The heat exchanger, and the scrubber systems are inspected at least annually for accumulation of uranium compounds. No significant accumulation has been observed over many years of operation. This discharge scrubber liquid is processed as described in Section 15.7.4.2.

Ash is removed from the incinerator via the vacuum collection hood, analyzed for total uranium, and dispositioned for burial or wet recovery.

Safety Features - Low level wastes are dispositioned for incineration after gamma counting. The wastes are logged in on the Incinerator/Scrubber Continuous Inventory Sheet and then subdivided into incinerator charges in the filter cut-up hood. Individual charges are packaged in plastic or paper bags.

The typical incinerator charge is less than 10 Kg of combustible waste and only a few grams of U-235. No significant ash accumulation has been observed in the secondary combustion chamber. Scrubbed gases are passed through a packed tower scrubber to remove any residual particulates before the effluent gases are discharged into the ventilation stack after the HEPA filter. This stack is continuously sampled.

Charging of the incinerator is terminated when the inventory sheet shows that a total of 800 g U-235 has been introduced into the system or when the ash has been removed. Normally, the ash requires removal prior to reaching the 800 g U-235 limit.

Pressure indicators are located before and after each stage of the system. Operating procedures require frequent checks of these indicators to assure that the entire system remains under negative pressure.

The gas firing system is provided with standard fire safety controls. Both burners have thermocouple controlled valves which close in the event the flame goes out. The valves will not open if the pilot light is out. Gas supply is cut off automatically if there is an electric power failure. The incinerator room is a separate fire control area with cinder block walls and has a fire door for entrance to Building 253.

There are no liquid or particulate discharges to the environs from the system. The used scrubber solution is evaporated to recover the solids contents.

Nuclear Safety - As noted above, a continuous inventory process is utilized to monitor the number of charges ($\leq$800 g U-235) incinerated in a given incineration campaign. An incineration campaign is terminated either by input grams U-235 ($\leq$ 800) or when the ash is removed. When all material dispositioned for incineration is incinerated, the incinerator is cleaned. Thus, the operation of the incinerator is predicated on a very conservative operational limit of 800 gr U-235, or less.

15.7.1.3 Sock and Reactor Metal Filter Cleaning

Description - Most sock filters are cleaned in-place by pulsed air blowback; these filters are periodically inspected for integrity and replaced, when necessary. Failed sock filters are cleaned in a hood and dispositioned for incineration.

Metal filters in the backup filter system of the R-2 and R-3 reactors undergo a cleaning in place by a process which diverts the particulate matter to a safe volume hopper. The removed material is dispositioned to a recycle process. Alternatively, these filters may be removed, placed in a plastic bag, and transferred to a utility hood for additional cleaning. The material is collected in a safe volume container and subsequently dispositioned a recycle process.

Safety Features - The in-place cleaning of the metallic filters for the R-2 and R-3 backup filters is accomplished by blowdown of the filter with nitrogen gas. During blowdown of the filter, the nitrogen gas and hydrogen fluoride pass through a sodium carbonate splash tank to remove the hydrogen fluoride. All filters are enclosed in plastic bags during transport and handling outside of filter housings and hoods. All hoods employ sufficient ventilation to assure an air velocity of at least 100 fpm on access openings.

Nuclear Safety - All containers employed to collect material removed from filters are safe volume containers. During transport and storage, if necessary prior to batching for recycle, these containers are appropriately labeled, closed, and spaced a minimum of one foot from each other and other SNM. Temporary storage areas for multiple containers employ rings in the floor to assure minimum spacing requirements.