7590-01

U. S. NUCLEAR REGULATORY COMMISSION

DOCKET 70-3070

LOUISIANA ENERGY SERVICES, L.P.

NOTICE OF AVAILABILITY OF THE

SAFETY EVALUATION REPORT

FOR THE CLAIBORNE ENRICHMENT CENTER

HOMER, LOUISIANA

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has published a Safety Evaluation Report (SER) (NUREG-1491) regarding the proposed construction and operation of the Claiborne Enrichment Center to be located near Homer, Louisiana. This report documents the Commission staff review and safety evaluation of the Louisiana Energy Services, L.P. (LES) application for a license to possess and use byproduct, source, and special nuclear material and to enrich natural uranium to a maximum of 5 percent U-235 by the gas centrifuge process.

The SER is available for public inspection and copying at the Commission's Public Document Room at the Gelman Building, 2120 L Street NW, Washington, DC and the Local Public Document Room at the Claiborne Parish Library, 901 Edgewood Drive, Homer, Louisiana. A free single copy of NUREG-1491 may be requested by writing to the Director, Division of Information Support Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

FOR FURTHER INFORMATION, CONTACT: Lidia A. Rochè, Enrichment Branch, Division of Fuel Cycle Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 504-2695.

Dated at Rockville, Maryland, this 24 day of January 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

John W. N. Hickey, Chief
Enrichment Branch
Division of Fuel Cycle Safety
and Safeguards
Office of Nuclear Material Safety
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NUREG–1491

Safety Evaluation Report
for the Claiborne Enrichment Center, Homer, Louisiana

Docket No. 70–3070

Louisiana Energy Services, L.P.

U.S. Nuclear Regulatory Commission

Office of Nuclear Material Safety and Safeguards

January 1994

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3. The National Technical Information Service, Springfield, VA 22161

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Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

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NUREG–1491

Safety Evaluation Report for the Claiborne Enrichment Center, Homer, Louisiana

Docket No. 70–3070

Louisiana Energy Services, L.P.

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NUREG–1491

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Docket No. 70–3070

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U.S. Nuclear Regulatory Commission

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January 1994

ABSTRACT

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff review and safety evaluation of the Louisiana Energy Services, L.P. (LES, the applicant) application for a license to possess and use byproduct, source, and special nuclear material and to enrich natural uranium to a maximum of 5 percent U-235 by the gas centrifuge process. The plant, to be known as the Claiborne Enrichment Center (CEC), would be constructed near the town of Homer in Claiborne Parish, Louisiana. At full production in a given year, the plant will receive approximately 4,700 tonnes of feed UF_6 and produce 870 tonnes of low-enriched UF_6, and 3,830 tonnes of depleted UF_6 tails. Facility construction, operation, and decommissioning are expected to last 5, 30, and 7 years, respectively.

The objective of the review is to evaluate the potential adverse impacts of operation of the facility on worker and public health and safety under both normal operating and accident conditions. The review also considers the management organization, administrative programs, and financial qualifications provided to assure safe design and operation of the facility. The NRC staff concludes that the applicant's descriptions, specifications, and analyses provide an adequate basis for safety review of facility operations and that construction and operation of the facility does not pose an undue risk to public health and safety.

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EXECUTIVE SUMMARY

Background

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff review and safety evaluation of the Louisiana Energy Services, L.P. (LES, the applicant) application for a license to possess and use byproduct, source, and special nuclear material and to enrich natural uranium to a maximum of five percent U-235 by the gas centrifuge process. The plant, to be known as the Claiborne Enrichment Center (CEC), would be constructed near the town of Homer in Claiborne Parish, Louisiana. The Application (LES, 1993k), Environmental Report (LES, 1993b), Safety Analysis Report (LES, 1993a), Proposed License Conditions (LES, 1993e and LES, 1994), and nonproprietary supporting information and communications are available at the NRC Public Document Room (2120 L Street, Washington DC) and at the Local Public Document Room (Claiborne Public Library, 901 Edgewood Drive, Homer, LA) under Docket No. 70-3070.

The plant's design capacity is 1.5 million separative work units per year. At full production in a given year, the plant will receive approximately 4,700 tonnes of feed UF_6, and produce 870 tonnes of low-enriched UF_6 and 3,830 tonnes of depleted UF_6 tails. Facility construction, operation, and decontamination and decommissioning are expected to last 5, 30, and 7 years, respectively.

The application was filed on January 29, 1991, by LES, a corporation comprised of four general partners and seven limited partners. The four general partners are Urenco Investments, Inc., Claiborne Fuels L.P. (a subsidiary of Fluor Daniel, Inc.), Claiborne Energy Services, Inc. (a subsidiary of Duke Power Company), and Graystone Corporation (a subsidiary of Northern States Power Company).

In accord with Public Law 101.575, the Solar, Wind, and Geothermal Power Production Incentives Act of 1990 revision of the Atomic Energy Act of 1954, uranium enrichment facilities will be licensed subject to the provisions of the Atomic Energy Act pertaining to source material (SM) and special nuclear material (SNM) (NRC, 1992d). Therefore, the primary bases for review of the application are the regulations of Parts 40 (NRC, 1961) and 70 (NRC, 1956a) of Title 10 of the Code of Federal Regulations (CFR). In addition, by Commission Order (NRC, 1991b), the draft "General Design Criteria" for uranium enrichment, published in the Advance Notice of Proposed Rulemaking for 10 CFR Part 76 (ANPR) (NRC, 1988a) and other special standards and instructions shall be applied. The Commission Order specifies that for the purpose of siting and design of a facility against accidental atmospheric releases of uranium hexafluoride (UF_6), health and safety criteria contained in NUREG-1391, "Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation" (NRC, 1991a), shall be applied. The criteria include a limiting intake of uranium in soluble form of 10 milligrams and a limiting exposure to hydrogen fluoride (HF) at a concentration of 25 milligrams per cubic meter for 30 minutes. Other regulations which shall be applied according to their terms include: 10 CFR Parts 19 (NRC, 1973), 20 (NRC, 1991c), 21 (NRC, 1977c), 30 (NRC, 1991d), 61 (NRC, 1992a), and 140 (NRC, 1960).

The objective of the review is to evaluate the potential adverse impacts of operation of the facility on worker and public health and safety under both normal operating and accident conditions. The review also considers the management organization, administrative programs, and financial qualifications provided to assure safe design and operation of the facility. The review followed the ANPR framework in identifying and evaluating those elements of plant design and operation, termed important to safety, which must function at the highest level of reliability. The function of these and related systems was evaluated for response to design basis events. Particular attention is given to criticality safety, which is evaluated in its administrative, design, and operational aspects. Normal operational impacts are assessed for maximally exposed individuals and for the surrounding population. The potential consequences of a set of accidents are estimated to identify the range of potential adverse impacts and to identify required limits for operation. Where the applicant's design or procedures should be supplemented, the NRC staff has recommended license conditions to provide additional assurance of safe operation. The following sections provide summaries of relevant site and facility characteristics and summarize the results and conclusions of the NRC staff safety evaluation.

Site Description

The site is located in Claiborne Parish, northwest Louisiana, approximately 8 kilometers (5 miles) northeast of the town of Homer, LA and 80 kilometers (50 miles) east-northeast of Shreveport, LA (LES, 1993a, 1993b). It consists of 179 hectares (442 acres) of land, approximately 28 hectares (70 acres) of which will be developed for plant facilities. Topography of the area around the CEC is characterized by rolling hills, with ground elevations ranging from 60 to 104 meters (200 to 340 feet) above sea level. The site is in the Ouachita River drainage basin with small creeks carrying water from the site to the east and west.

The population of Claiborne Parish was approximately 17,400 in 1990 and the population within 80 kilometers (50 miles) of the site was approximately 349,000. Most of the population of the parish live in the towns of Homer and Haynesville. The permanent residence nearest to the site is located approximately 475 meters (0.3 miles) north-northeast of the plant stacks.

Industrial facilities in the vicinity of the site include oil and gas wells and manufacturing businesses. Claiborne Parish has approximately 1,400 producing wells, and the total number of employees of manufacturing concerns is estimated to be less than 700. The NRC staff concludes that the surrounding facilities do not pose a risk to operation of the CEC.

The nuclear facility nearest to the CEC is the Grand Gulf Nuclear Power Station in Port Gibson, MS, approximately 215 kilometers (135 miles) south of the CEC site. The NRC staff concludes that no interactions of these facilities would occur and that doses in the CEC area from the other nuclear facilities would be insignificant.

The climate of north-central Louisiana is transitional between the subtropical, humid climate of the Gulf of Mexico and the continental climates of the great plains and midwest. The average annual temperature is 18.6 °C (65.4 °F), and average annual rainfall is approximately 130 centimeters. High winds in the vicinity of the site are most frequently associated with thunderstorms and, due to the rolling hills and forest cover, tornadoes are not common. The applicant has provided a probabilistic assessment of the high wind and tornado hazard for the site. The ANPR specified a frequency of occurrence of $1x10^{-4}$ per year as the design basis for enrichment facilities. Applicant analysis reported that at this frequency, the design basis tornado (DBT) has a wind speed of 51.4 m/s (115 mph) and an atmospheric pressure change of 1915 pascals (40 psf). The NRC staff has reviewed the applicant analysis in light of NRC guidance and concludes that the results provide an acceptable design basis for the CEC.

The applicant has based analysis of atmospheric dispersion at the site on 5 years of meteorological data collected at Shreveport, LA. In order to establish the validity of this approach, the Shreveport data was compared with data collected for two other north-Louisiana sites. The NRC staff has reviewed the applicant analysis and concludes that the Shreveport data is likely to be representative of meteorological conditions at the site. Joint frequencies of concentration per unit source (χ/Q) estimated for normal operational releases show a maximum annual average value of $5.5x10^{-7}$ s/m^3 at a location 800 meters (0.5 miles) north of the plant stacks.

The applicant has provided an assessment of flooding potential at the site based on Corp of Engineer (COE) analysis and U.S. Geologic Survey maps. The site is not located near the floodplain of a major river and consequently the site design basis flood results from local intense precipitation. The applicant estimated that up to 6.2 centimeters (2.5 inches) of water could pond on the site during the design basis storm. The NRC staff has reviewed the applicant's analysis and concludes that the results are an acceptable design basis for the CEC.

The geology of northern Louisiana reflects the deposition of sedimentary rocks throughout the Cenozoic and Mesozoic Eras with Triassic basement rock located approximately 6,000 meters (20,000 feet) below the ground surface. The site is located within the Interior Salt Basin seismotectonic region. The region has historically experienced minimal seismicity and is generally considered aseismic. The largest recorded earthquake in the vicinity of the CEC site was a magnitude 4.6 earthquake at a distance of 169 kilometers (105 miles) in 1911. In order to fulfill the ANPR requirement that the facility design basis earthquake (DBE) has a return period of 500 years, the applicant performed a probabilistic assessment of seismic hazard. The procedure included identification of seismic zones, analysis of historical earthquakes and related faulting, and attenuation of earthquake effects from the source to the site. The design basis earthquake identified by the applicant procedure was characterized by a horizontal acceleration at bedrock of 0.046g. The NRC staff concludes that the method is state-of-the-art and acceptable and that the results fulfill the requirements of the ANPR. In addition to this analysis, the applicant developed horizontal and vertical response spectra for the site. The NRC staff concludes that the results for the horizontal response spectra are acceptable but that the vertical response spectra may not be conservative. Thus, the NRC staff concludes that the applicant's analysis of geology and seismology in conjunction with independent NRC staff analysis based on the Southern Building Code Congress International (SBCCI) Standard Building Code and Regulatory Guide 1.60 procedures, which do not require reliance on the applicant's vertical response spectra, provides an acceptable design basis for the CEC.

Plant Description

The CEC is a process plant designed to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream enriched in the uranium-235 isotope and a tails stream depleted in the uranium-235 isotope. The process, entirely physical in nature, takes advantage of the tendency of materials of differing density to segregate in the force field produced by a centrifuge. The chemical form of the working material of the plant, uranium hexafluoride (UF_6), does not require chemical transformations at any stage of the process. Solid UF_6 is delivered to the plant in cylinders containing up to 12.6 tonnes (14 tons) of UF_6. Feed cylinders received at the plant are inspected and weighed in the Cylinder Receipt and Dispatch Building (CRDP) and transferred to the main process facility, the Separations Building. Separation operations are divided among three Plant Units, each capable of handling one-third of plant capacity. Each Plant Unit is divided into two Assay Units, and each assay unit is comprised of 7 cascades. Therefore, the total plant is comprised of 42 cascades. Each cascade is comprised of approximately 1,000 centrifuges.

The enrichment process housed in the Separations Building is comprised of four major elements: a feed system, an enrichment system, a product take-off system, and a tails take-off system. Support functions include product sampling and blending systems and analytic and decontamination systems. The major pieces of equipment used in the feed process are autoclaves, desublimers, and take-up cylinder cubicles. Feed cylinders are loaded into electrically heated autoclaves; vented for removal of light gases, primarily air and HF; and heated to liquefy and vaporize the UF_6. The light gases and UF_6 vapor generated during cylinder venting are routed to the desublimer (that is, a cold trap) where the UF_6 is desublimed. The light gases are routed to a process gas clean-up system called the Gaseous Effluent Vent System (GEVS). The GEVS has activated carbon adsorbent and High Efficiency Particulate Air (HEPA) filters which remove most of the HF and uranium compounds before the gas is released to the atmosphere. The UF_6 solidified in the desublimer is sublimed and transferred to a cylinder in the take-up cubicle for re-use as feed material. After venting, UF_6 from the feed autoclave is routed to the separation cascades. Pressure in all process lines outside of the autoclaves is subatmospheric.

Gaseous UF_6 from the feed autoclaves is routed to the centrifuge cascades. Each centrifuge is a thin-walled, vertical, cylindrically shaped rotor which spins around a central post within an outer casing. Feed, product, and tails streams enter and leave the centrifuge through a central post. Control valves, resistor orifices, and controllers provide uniform flow of product and tails.

Depleted UF_6 exiting the cascades is compressed from the high vacuum of the centrifuge to approximately 23,000 pascals (3 psia) for desublimation into tails cylinders. The primary equipment of the tails take-off system is the vacuum pumps and the take-off cylinder stations. Cooling water is sprayed over cylinders in the take-off stations to effect the desublimation. Filling of the cylinders is monitored with a load cell system, and filled cylinders are transferred to an outdoor storage area after solidification of the UF_6.

Enriched UF_6 from the cascades is desublimed in a product take-off system comprised of vacuum pumps, product cylinder take-off stations, and a desublimer. The pumps compress the UF_6 from the low pressure of the centrifuge to approximately 45,000 pascals (6 psia). The heat of desublimation of the UF_6 is removed by cooling air routed through the product cylinder take-off station. The product stream contains any light gases which may have passed through the centrifuge cascades. Therefore, a desublimer is provided to vent these gases from the product cylinder. Any UF_6 vented to the desublimer is transferred to another product cylinder for use as product or blending stock. Filling of the cylinders is monitored with a load cell system, and filled cylinders are transferred to product liquid sampling autoclaves for assay.

The sampling autoclave is an electrically heated, closed pressure vessel used to liquefy the UF_6 and allow collection of a sample. The autoclave is fitted with a hydraulic tilting mechanism which elevates one end of the autoclave so that liquid UF_6 pours into a sampling manifold connected to the cylinder valve. After sampling, the cylinder is indirectly cooled by water flowing through coils located in the autoclave jacket.

LES customers may require product at enrichment levels other than that produced by a single CEC Assay Unit. Therefore, the plant has the capability to blend enriched UF_6 from two donor cylinders into a product receiver cylinder. The blending system is comprised of autoclaves for the two donor cylinders and a take-up station for the receiver cylinder. Product UF_6 is desublimed in the take-up station using recirculated cooling air.

Support functions, including sample analysis and equipment decontamination, are conducted in the Technical Services Area (TSA) of the Separations Building. Decontamination, primarily of pumps and valves, uses solutions of citric acid. Storage areas and portions of the liquid and solid waste management systems are also located in the TSA.

Design of Structures, Systems, and Components

The NRC staff applied a structured release scenario development and impact analysis procedure to identify structures, systems, and components important to safety. The NUREG-1391 criteria for uranium intake and HF concentration were used to determine limiting release quantities of UF_6. Individual elements of the plant were examined for single active failures and response to design basis events. Potential consequences of release scenarios developed in this manner were compared with the limiting release quantities to identify structures, systems, and components important to safety. The NRC staff analysis identified heater protection circuits (temperature and pressure) of the feed, sampling, and blending autoclaves as important to safety.

Response to Design Basis Events

Plant systems important to safety or systems whose operation might affect operation of systems important to safety must maintain function given the occurrence of ANPR-specified design basis events. The NRC staff determined that the design basis flood does not affect plant systems. The Separations Building comprises cylindrical steel stacks, rectangular concrete columns, solid concrete walls, precast/prestressed concrete beams, and double-tee roof and floor members. For response to the DBT, the applicant provided analysis to identify required load combinations and to demonstrate that roof members had required grout, and that walls and stacks had adequate thickness to resist DBT forces and missiles. For the DBE, the applicant identified required load combinations, calculated resultant member forces, and compared these forces to the allowable member limits. The NRC staff reviewed the applicant's analysis and performed supplementary analysis to conclude that the Separations Building would maintain its physical integrity in the design basis earthquake and tornado.

The NRC staff reviewed the applicant's analysis and performed supplementary calculations to determine the response of the autoclaves and the autoclave heater protection controls to the design basis earthquake. The Separations Building protects components inside the building from the effects of the DBT. For the feed and blending autoclaves, the applicant provided a design including force and moment balances for equipment to be built to American Institute of Steel Constructors (AISC) and American Society of Mechanical Engineers (ASME) code specifications. The NRC staff concludes that autoclaves built to this design will maintain function during the DBE. For the liquid sampling autoclave, the applicant provided a design specification which was reviewed by the NRC staff. The NRC staff concludes that a product liquid sampling autoclave built to this specification would maintain function during the DBE but proposes an inspection license condition to confirm that the autoclave is built to the specification. For autoclave foundations and Class I electrical and control systems, the applicant provided design calculations based on force and moment balances. The NRC staff reviewed the applicant's design calculations, performed supplementary analysis, and concludes that the autoclave foundations and Class I controls will maintain function during the DBE.

Instruments and Controls

Instrumentation and controls at the CEC are used to direct normal operations and to protect against abnormal events or accidental releases. The upper-level approach is to have the primary control and protection functions performed at local control centers (LCC) near the equipment and to have all control functions duplicated in a central control room (CCR). Each piece of equipment is controlled from an LCC which has state switches which provide for selection of valve positions and interlocks for allowed operating states of that equipment. Generally, controllers for individual elements have three set points. The lower two levels are used to maintain process variables within specified ranges, while the upper set point is used to shut down the controlled element. Monitoring of the confinement barrier is provided primarily by measurement of pressure. Elevated pressure indicates an abnormal condition with potential inleakage of air or exceedance of specified temperature. For each autoclave, redundant autoclave air space temperature and pressure monitors are used to shut down the heaters if control limits are exceeded. The ANPR specifies criteria for control room instrumentation, for performance of controls with safety significance, and for performance of controls for process operation and shutdown. The NRC staff reviewed the design of CEC instrumentation and control systems and concludes that the system designs meet the ANPR requirements.

Waste Management

Gaseous, liquid, and solid waste will be produced and managed during operation of the CEC. Potentially contaminated wastes streams are released to the environment from the GEVS and a portion of the TSA ventilation system. All contaminated gaseous effluents pass through stack monitors prior to release to the atmosphere. Gas exiting the GEVS has passed through carbon, alumina, and High Efficiency Particulate Air (HEPA) filters for removal of contaminants. Gas exiting the potentially contaminated portion of the TSA HVAC system passes through a HEPA filter prior to release to the atmosphere. The average annual release of uranium to the atmosphere is expected to be less than $4.4x10^{+6}$ Bq (120 μCi). For this release rate, the NRC staff estimates that the uranium concentration at the point of maximum exposure would be $7.7x10^{-8}$ Bq/m^3 ($2.0x10^{-18}$ μCi/ml). This concentration is six orders of magnitude less than the 10 CFR Part 20 limit for releases to unrestricted areas. The average release rate of HF, the hazardous chemical of concern at the CEC, is expected to be less than 6.5 kilograms/yr. The NRC staff estimates that the HF concentration at the point of maximum exposure is $1.1x10^{-7}$ mg/m^3. The estimate is a factor of 30 million less than the American Conference of Governmental Industrial Hygienists' Time Weighted Average limit for occupational exposures.

Liquids contaminated with uranium which are generated in CEC operations would pass through a Liquid Waste Disposal System (LWDS) prior to release to Bluegill Pond through the Sewage Treatment System. The LWDS uses precipitation, evaporation, and ion exchange to remove uranium isotopes from the liquid stream. Contaminated solids generated in the LWDS would be disposed offsite at an authorized facility. For the purpose of impact analysis, the NRC staff adopted $1.0x10^{+6}$ Bq/yr (28 μ Ci/yr) as a conservative estimate of the upper limit for the annual release rate of uranium isotopes to Bluegill Pond. LWDS effluent is sampled and analyzed prior to release to the Sewage Treatment System and Sewage Treatment System effluent is sampled during release to Bluegill Pond. The action level for alpha activity in the liquid stream released to Bluegill Pond corresponds to 0.5 percent of the 10 CFR Part 20 limit for release to unrestricted areas.

Solid wastes generated at the CEC include radioactive, hazardous, mixed, and industrial wastes. Less than 100 kilograms of uranium, nearly all in the LWDS evaporator bottoms precipitate and GEVS carbon adsorbent, would be generated annually. All solid wastes would be disposed offsite at authorized facilities.

The NRC staff reviewed the design of CEC systems for management of radioactive, hazardous, mixed, and industrial waste. The review established that CEC radioactive waste management systems use effective process components which would limit releases to small fractions of 10 CFR Part 20 limits. The NRC staff concludes that atmospheric and liquid effluents would be controlled to levels as low as is reasonably achievable. The NRC staff concludes that the effluent monitoring systems have acceptable levels of detection and would protect public health and safety. Systems for control of mixed, hazardous, and industrial wastes will be managed subject to applicable regulations.

Radiation Protection for Workers and the Public

Radiation protection for CEC workers is provided by a radiation protection program that uses facility design features, designation of potentially contaminated areas, appropriate work control procedures, continuous air monitoring, area surveys, protective clothing, and worker dose assessments to provide a safe working environment. All personnel entering radiation control areas will wear external radiation monitoring dosimeters and internal doses will be assessed in a bioassay program. Alpha-in-air monitors are provided at more than 40 locations throughout the Separations Building to detect the presence of uranium in the air.

The NRC staff, using the average annual release rates of uranium to air and water, assessed the potential radiological impacts to the public of normal operations. For the air pathway, the annual dose for the critical individual is estimated to be 2.4×10^{-9} Sv (2.4×10^{-7} rem), and the population dose is estimated to be 2.8×10^{-6} person-Sv (2.8×10^{-4} person-rem). For the liquid pathway, the annual dose for the critical individual is estimated to be 6.0×10^{-6} Sv (6.0×10^{-4} rem), and the population dose is estimated to be 4.9×10^{-2} person-Sv (4.9 person-rem). An operational environmental monitoring program will provide additional assurance that public exposure to radiation is within allowable limits.

The NRC staff reviewed proposed CEC radiation protection programs against the ANPR and regulatory requirements. The NRC staff estimates of potential occupational and public exposure to radioactivity were small, well below regulatory limits, and as low as is reasonably achievable. The NRC staff reviewed the radiation protection programs proposed for the CEC and concludes that they would protect worker and public health and safety and are therefore acceptable.

Criticality Safety in Design and Operation

The NRC staff has reviewed the applicant's program for administration of criticality safety and the proposed nuclear criticality safety factors for design and operation. The NRC staff review of the proposed administrative practices found the program acceptable. The NRC staff reviewed the applicant's analysis of safety factors for mass/geometric units and compared this analysis with Regulatory Guide 3.52 safety margins. The NRC staff concludes that the proposed criticality safety criteria are adequate to provide reasonable assurance that worker and public health and safety are protected. The NRC staff reviewed the proposed design for individual pieces of equipment against criticality safety criteria and concludes that the plant can be safely operated.

Management Systems and Controls

The NRC staff has reviewed the management structure and administrative programs proposed for the CEC. The NRC staff evaluated the management organization proposed for the CEC and the qualifications proposed for the various managerial functions. The NRC staff evaluated programs proposed for training; development, modification, and management of procedures; conduct of operational and safety audits; operation of safety committees; maintenance of records; and emergency planning. The NRC staff concludes that managerial structure and administrative programs meet all regulatory requirements and provide for safe operation of the CEC.

Accident Analysis

The NRC staff reviewed the accident analysis presented in the SAR and performed independent analysis of a set of potential accidents related to operation of the CEC. The estimated consequences of the potential accidents ranged in severity from minimal onsite consequences to detectable offsite consequences. In developing the representative set of potential accidents, the NRC staff considered past experience at fuel cycle facilities handling UF_6, experience at Urenco facilities in Europe, and the CEC design described in the SAR. The only accident deemed to be possible at the CEC which could have significant offsite consequences is release of a large quantity of UF_6 from a cylinder containing liquid UF_6. Under normal operating conditions, UF_6 is present in the liquid state only in cylinders inside of autoclaves. Attainment of the elevated temperature and pressure required to breach a cylinder and autoclave is prevented by heater shutoff circuits activated by occurrence of high temperature or pressure. These protection circuits are provided for each autoclave. The protection circuits are designated Class I, important to safety, and are designed to maintain function in the occurrence of design basis events. The severity of the design basis events is specified in the ANPR. Uncontrolled fire is an additional source of energy which could possibly lead to rupture of a UF_6 cylinder. The CEC design precludes occurrence of this scenario through absence of combustible material in areas where significant quantities of UF_6 are present and limitation on fuel inventories on UF_6 cylinder transporters. Due to the design features and the use of redundant, diverse protection circuits, the NRC staff concludes that occurrence of potential accidents at the CEC does not pose an undue risk to public health and safety.

Quality Assurance

The ANPR specifies that Quality Assurance (QA) programs applied for structures, systems, and components important to safety meet the requirements of 10 CFR Part 50, Appendix B. The applicant has proposed a graded, three-level QA program which follows the guidelines of ASME NQA-1 for structures, systems, and components important to safety and intermediate levels of controls for less sensitive systems. The NRC staff has reviewed the QA program proposed for design and construction, and start-up and operation and concludes that the program will meet the requirements of 10 CFR Part 50, Appendix B, and is therefore acceptable.

Financial Qualifications

Four general and seven limited partners plan to construct, operate, and decommission the CEC. The applicant estimates that hard construction costs will be $816 million in 1992 dollars. Approximately 30 percent equity financing will be used. The NRC staff reviewed the financial status of each of the partners, including reports of shareholder equity, cash flow, and cash on hand to evaluate the source of and reliability of funds. The applicant commits to maintaining $200 million in nuclear liability insurance. The NRC staff reviewed the applicant's estimate of product market prices and considered project risk. On the basis of this review, the NRC staff concludes that the financial risk of the project should not affect the protection of public health and safety.

Safeguards and Security

The NRC staff has reviewed the applicant's Fundamental Nuclear Material Control (FNMC) Plan against the requirements of 10 CFR Part 74 and supporting NRC guidance. The FNMC Plan includes descriptions of performance objectives and system capabilities, including the means for precluding or detecting unauthorized enrichment activities. On the basis of this review, the NRC staff concludes that the applicant's proposed FNMC Plan, when implemented, is acceptable for meeting the requirements of 10 CFR 74.33.

The NRC staff has reviewed the applicant's Physical Security Plan against the requirements of 10 CFR Part 73 and supporting NRC guidance. The applicant's plan includes constructing the facility within a controlled area surrounded by a chain link fence. The controlled area is further surrounded by a cleared area, and access is monitored and controlled by watchmen. Security patrols and communications are provided as a response to unauthorized penetrations of the controlled area. Personnel working at the facility will be screened for trustworthiness. Notification, confirmation, and inspections are proposed to control special nuclear material shipments. On the basis of this review, the NRC staff concludes that the applicant's Physical Security Plan meets the requirements of 10 CFR 73.67 and is thus acceptable.

The applicant is required to use, process, store, reproduce, transmit, or handle National Security Information (NSI) and/or Restricted Data (RD) in accordance with 10 CFR Part 95.

The NRC staff has reviewed and approved the applicant's plan, which is not releasable to the public, for control and protection of NSI and RD.

Decontamination and Decommissioning

Facilities licensed by the NRC under 10 CFR Parts 40 and 70 are decommissioned by the licensee in order to permit release of the site and facilities for unrestricted use and to terminate the license. In order to decontaminate and decommission (D&D) the facility, the applicant proposes to incorporate specific features into the design which will facilitate D&D, characterize the facility and site after termination of enrichment operations, prepare a detailed decommissioning plan, complete D&D activities, and complete a final site survey. The entire D&D process is estimated to require 7 years and will involve installation of two new facilities within existing buildings for decontamination operations. The applicant's estimate of D&D cost is approximately $518 million, including $485 million estimated for dispositioning of depleted uranium tails. The NRC staff concludes that the applicant's proposed procedures and funding are adequate to D&D the facility. In addition, the applicant has committed by license condition to reviewing and updating the decommissioning funding plan at least every five years, starting from the time of issuance of the license.

Conclusions

The NRC staff has reviewed the applicant's SAR, Proposed License Conditions, and supporting documentation, including responses to NRC requests for additional information, and concludes that the applicant's descriptions, specifications, and analyses provide an adequate basis for safety review of facility operations. Further, the NRC staff concludes, on the basis of the NRC staff review of applicant's submissions and independent NRC staff analyses as summarized above, that construction and operation of the facility does not pose an undue risk to public health and safety. In order to provide additional assurance that the bases for this conclusion remain unchanged and in accord with Public Law 101.575, NRC staff will perform a preoperational inspection to confirm that the construction and installation of each plant unit is in accordance with the requirements of the license.

1 INTRODUCTION AND GENERAL DESCRIPTION

1.1 Introduction

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff safety review and evaluation of the Louisiana Energy Services, L.P. (LES, the applicant) application for a license to possess and use byproduct, source, and special nuclear material and to enrich natural uranium to a maximum of 5 percent U-235 by the gas centrifuge process. The plant, to be known as the Claiborne Enrichment Center (CEC), would be constructed near the town of Homer in Claiborne Parish, Louisiana. The License Application (LES, 1993k), Environmental Report (ER) (LES, 1993b), Safety Analysis Report (SAR) (LES, 1993a), Proposed License Conditions (PLC) (LES, 1993e and LES, 1994), and nonproprietary supporting information and communications are available at the NRC Public Document Room (2120 L Street, N.W., Washington DC) and at the Local Public Document Room (Claiborne Parish Library, 901 Edgewood Drive, Homer, LA) under Docket No. 70-3070.

The plant design capacity is 1.5 million Separative Work Units (SWU) per year. At full production in a given year, the plant will receive approximately 4,700 tonnes of feed uranium hexafluoride (UF_6), and produce 870 tonnes of low enriched UF_6 and 3,830 tonnes of depleted UF_6 tails. Facility construction, operation, and decontamination and decommissioning are expected to last five, thirty, and seven years, respectively.

The application was filed on January 29, 1991 by LES, a corporation comprised of four general partners and seven limited partners. The four general partners are Urenco Investments, Inc., Claiborne Fuels L.P. (a subsidiary of Fluor Daniel, Inc.), Claiborne Energy Services, Inc. (a subsidiary of Duke Power Company), and Graystone Corporation (a subsidiary of Northern States Power Company).

In accord with Public Law 101.575, the Solar, Wind, and Geothermal Power Production Incentives Act of 1990 revision of the Atomic Energy Act of 1954, uranium enrichment facilities will be licensed subject to the provisions of the Atomic Energy Act pertaining to source material (SM) and special nuclear material (SNM) (NRC, 1992d). Therefore, the primary bases for review of the application are the regulations of Parts 40 (NRC, 1961) and 70 (NRC, 1956a) of Title 10 of the Code of Federal Regulations (CFR). In addition, by Commission Order (NRC, 1991b), the draft "General Design Criteria" for uranium enrichment, published in the Advance Notice of Proposed Rulemaking for 10 CFR Part 76 (ANPR) (NRC, 1988a), and other special standards and instructions shall be applied. The Commission Order specifies that for the purpose of siting and design of a facility against accidental atmospheric releases of uranium hexafluoride (UF_6), health and safety criteria contained in NUREG-1391, "Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation" (NRC, 1991a) shall be applied. The criteria include a limiting intake of uranium in soluble form of 10 milligrams and a limiting exposure to hydrogen fluoride (HF) at a concentration of 25 milligrams per cubic meter for 30 minutes. Other regulations which shall be applied according to their terms include: 10 CFR Parts 19 (NRC, 1973), 20 (NRC, 1991c), 21 (NRC, 1977c), 30 (NRC, 1991d), 61 (NRC, 1992a), and 140 (NRC, 1960).

The objective of the review is to evaluate the potential adverse impacts of operation of the facility on worker and public health and safety under both normal operating and accident conditions. The review also considers the management organization, administrative programs, and financial qualifications provided to assure safe design and operation of the facility. The review followed the ANPR framework in identifying and evaluating those elements of plant design and operation, termed important to safety, which must function at the highest level of reliability. The function of these and related systems was evaluated for response to design basis events. Particular attention is given to criticality safety, which is evaluated in its administrative, design, and operational aspects. Normal operational impacts are assessed for maximally exposed individuals and for the surrounding population. The potential consequences of a set of accidents are estimated to identify the range of potential adverse impacts and to identify required limits for operation. Where the applicant's design or procedures should be supplemented, the NRC staff has recommended license conditions to provide additional assurance of safe operation. The following sections provide summaries of relevant site and facility characteristics and summarize the results and conclusions of the safety evaluation for the principal review matters.

1.2 General Plant Description

1.2.1 Site Description

The site is located in Claiborne Parish, northwest Louisiana, approximately 8 kilometers (5 miles) northeast of the town of Homer, LA and 80 kilometers (50 miles) east-northeast of Shreveport, LA (LES, 1993a, 1993b). It consists of 179 hectares (442 acres) of land, approximately 28 hectares (70 acres) of which will be developed for plant facilities. Topography of the area around the CEC is characterized by rolling hills, with ground elevations ranging from 60 to 104 meters (200 to 340 feet) above sea level. The site is in the Ouachita River drainage basin with small creeks carrying water from the site to the east and west.

1.2.2 Plant Description

The CEC is a process plant designed to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream enriched in the uranium-235 isotope and a tails stream depleted in the uranium-235 isotope. The process, entirely physical in nature, takes advantage of the tendency of materials of differing density to segregate in the force field produced by a centrifuge. The chemical form of the working material of the plant, UF_6, does not require chemical transformations at any stage of the process. Solid UF_6 is delivered to the plant in cylinders containing up to 12.6 tonnes (14 tons) of UF_6. Feed cylinders received at the plant are inspected and weighed in the Cylinder Receipt and

Dispatch Building (CRDP), and transferred to the main process facility, the Separations Building. Separation operations are divided among three Plant Units, each capable of handling one-third of plant capacity. Each Plant Unit is divided into two Assay Units and each assay unit is comprised of 7 cascades. Therefore, the total plant is comprised of 42 cascades. Each cascade is comprised of approximately 1,000 centrifuges.

The enrichment process housed in the Separations Building is comprised of four major elements: a feed system, an enrichment system, a product take-off system, and a tails take-off system. Support functions include product sampling and blending systems and analytic and decontamination systems. The major pieces of equipment used in the feed process are autoclaves, desublimers, and take-up cylinder cubicles. Feed cylinders are loaded into electrically heated autoclaves; vented for removal of light gases, primarily air and HF; and heated to liquefy and vaporize the UF_6. The light gases and UF_6 vapor generated during cylinder venting are routed to the desublimer (that is, a cold trap) where the UF_6 is desublimed. The light gases are routed to a process gas clean-up system called the Gaseous Effluent Vent System (GEVS). The GEVS has activated carbon adsorbent and High Efficiency Particulate Air (HEPA) filters which remove most of the HF and uranium compounds before the gas is released to the atmosphere. The UF_6 solidified in the desublimer is sublimed and transferred to a cylinder in the take-up cubicle for re-use as feed material. After venting, UF_6 from the feed autoclave is routed to the separation cascades. Pressure in all process lines outside of the autoclaves is subatmospheric.

Gaseous UF_6 from the feed autoclaves is routed to the centrifuge cascades. Each centrifuge is a thin-walled, vertical, cylindrically shaped rotor which spins around a central post within an outer casing. Feed, product, and tails streams enter and leave the centrifuge through a central post. Control valves, resistor orifices, and controllers provide uniform flow of product and tails.

Depleted UF_6 exiting the cascades is compressed from the high vacuum of the centrifuge to approximately 23,000 pascals (3 psia) for desublimation into tails cylinders. The primary equipment of the tails take-off system is the vacuum pumps and the take-off cylinder stations. Cooling water is sprayed over cylinders in the take-off stations to effect the desublimation. Filling of the cylinders is monitored with a load cell system, and filled cylinders are transferred to an outdoors storage area after solidification of the UF_6.

Enriched UF_6 from the cascades is desublimed in a product take-off system comprised of vacuum pumps, product cylinder take-off stations, and a desublimer. The pumps compress the UF_6 from the low pressure of the centrifuge to approximately 45,000 pascals (6 psia). The heat of desublimation of the UF_6 is removed by cooling air routed through the product cylinder take-off station. The product stream contains any light gases which may have passed through the centrifuge cascades. Therefore, a desublimer is provided to vent these gases from the product cylinder. Any UF_6 vented to the desublimer is transferred to another product cylinder for use as product or blending stock. Filling of the cylinders is monitored with a load cell system, and filled cylinders are transferred to product liquid sampling autoclaves for assay.

The sampling autoclave is an electrically heated, closed pressure vessel used to liquefy the UF_6 and allow collection of a sample. The autoclave is fitted with a hydraulic tilting mechanism which elevates one end of the autoclave so that liquid UF_6 pours into a sampling manifold connected to the cylinder valve. After sampling, the cylinder is indirectly cooled by water flowing through coils located in the autoclave jacket.

LES customers may require product at enrichment levels other than that produced by a single CEC Assay Unit. Therefore, the plant has the capability to blend enriched UF_6 from two donor cylinders into a product receiver cylinder. The blending system is comprised of autoclaves for the two donor cylinders and a take-up station for the receiver cylinder. Product UF_6 is desublimed in the take-up station using recirculated cooling air.

Support functions, including sample analysis and equipment decontamination, are conducted in the Technical Services Area (TSA) of the Separations Building. Decontamination, primarily of pumps and valves, uses solutions of citric acid. Storage areas and portions of the liquid and solid waste management systems are also located in the TSA.

1.3 Identification of Agents and Contractors

LES, the applicant, is responsible for CEC design, quality assurance, construction, pre-operational testing, initial start-up, and operation. LES contracted with Urenco, a general partner, to provide the reference design for the facility. Urenco has experience in the gas centrifuge uranium enrichment process, operating three enrichment plants in Europe. LES contracted with Fluor Daniel and Duke Engineering Services to specify structures and facilities for the plant and to provide elements of the dose assessment and safety analyses. LES used consultants, Westinghouse Environmental and Geotechnical Services and Law Engineering Services, to provide site suitability and seismic analyses.

1.4 Summary of Review and Findings

Principal review matters for evaluation of the CEC include: description of site conditions and characterization of design basis natural phenomena; identification of structures, systems, and components important to safety; response of systems important to safety to design basis events; criticality safety in design and operation; function of instrumentation and controls; waste management; radiation protection of workers and the public; quality assurance; management systems and controls; accident analysis; financial qualifications; safeguards and security; and decontamination and decommissioning. The reviews and conclusions for these areas are summarized in the following paragraphs.

Site Characterization

Population density, land use, and physical characteristics of the site were reviewed against ANPR and regulatory requirements. The NRC staff determined that the applicant's analysis had identified appropriate characteristics for design basis natural phenomena including, earthquake, tornado, and flood. The NRC staff concludes that the characterization of the site provides an adequate basis for safety review.

Identification of Structures, Systems, and Components Important to Safety

The NRC staff applied a structured release scenario development and impact analysis procedure to identify structures, systems, and components important to safety. The NUREG-1391 criteria for uranium intake and HF concentration were used to determine limiting release quantities of UF_6. Individual elements of the plant were examined for single active failures and response to design basis events. Potential consequences of release scenarios developed in this manner were compared with the limiting release quantities to identify structures, systems, and components important to safety (Class I). The NRC staff analysis identified heater protection circuits (temperature and pressure) of the feed, sampling, and blending autoclaves as important to safety.

Response to Design Basis Events

Plant systems important to safety or systems whose operation might affect operation of important-to-safety systems must maintain function given the occurrence of ANPR-specified design basis events. The NRC staff determined that the design basis flood does not affect plant systems. The NRC staff reviewed the applicant's analysis and performed supplementary analysis to conclude that the Separations Building maintains its integrity in the design basis earthquake and tornado. The NRC staff reviewed applicant analysis and performed supplementary calculations to determine the response of the autoclaves and the autoclave heater protection controls to the design basis earthquake. The Separations Building protects components inside the building from the effects of the design basis tornado. The NRC staff concludes that the important-to-safety mechanical and control components built to the CEC design will survive the design basis earthquake.

Criticality Safety in Design and Operation

The NRC staff has reviewed the applicant's program for administration of criticality safety and the proposed nuclear criticality safety factors for design and operation. The NRC staff review of the proposed administrative practices found the program acceptable. The NRC staff reviewed the applicant's analysis of safety factors for mass/geometric units and compared this analysis with Regulatory Guide 3.52 safety margins. The NRC staff concludes that the proposed criticality safety criteria are adequate to provide reasonable assurance that worker and public health and safety are protected. The NRC staff reviewed the proposed design for individual pieces of equipment against criticality safety criteria and concludes that the plant can be safely operated.

Instruments and Controls

The ANPR specifies criteria for control room instrumentation, for performance of controls with safety significance, and for performance of controls for process operation and shutdown. The NRC staff reviewed the design of CEC instrumentation and control systems and concludes that the system designs meet the ANPR requirements.

Waste Management

The NRC staff reviewed the design of CEC systems for management of radioactive, hazardous, mixed, and industrial waste. The review established that CEC radioactive waste management systems use effective process components which would limit releases to small fractions of 10 CFR Part 20 limits. The NRC staff concludes that atmospheric and liquid effluents would be controlled to levels as low as reasonably achievable. The NRC staff concludes that the effluent monitoring systems have acceptable levels of detection and would protect public health and safety. Systems for control of mixed, hazardous, and industrial wastes are also in compliance with applicable regulations.

Radiation Protection for Workers and the Public

The NRC staff reviewed proposed CEC Radiation Protection programs against the ANPR and regulatory requirements. The NRC staff estimates of potential occupational and public exposure to radioactivity were small, well below regulatory limits, and as low as reasonably achievable. The NRC staff reviewed the radiation protection programs proposed for the CEC and concludes that they would protect worker and public health and safety and are therefore acceptable.

Quality Assurance

The ANPR specifies that Quality Assurance (QA) programs applied for structures, systems, and components important to safety meet the requirements of 10 CFR Part 50, Appendix B. The applicant has proposed a graded, three-level QA program which follows the guidelines of ASME NQA-1 for structures, systems, and components important to safety and intermediate levels of controls for less sensitive systems. The NRC staff has reviewed the QA program proposed for design and construction, and start-up and operation and concludes that the program will meet the requirements of 10 CFR Part 50, Appendix B and is therefore acceptable.

Management Systems and Controls

The NRC staff has reviewed the management structure and administrative programs proposed for the CEC. The NRC staff evaluated the management organization proposed for the CEC and the qualifications proposed for the various managerial functions. The NRC staff evaluated programs proposed for training, development and modification of procedures, conduct of operational and safety audits, operation of radiation protection and safety committees, conduct of maintenance, emergency planning, and inspection of start-up and operations. The NRC staff concludes that managerial structure and administrative programs meet all regulatory requirements and provide for safe operation of the CEC.

Accident Analysis

The NRC staff reviewed the accident analysis presented in the SAR and performed independent analysis of a set of potential accidents related to operation of the CEC. The estimated consequences of the potential accidents ranged in severity from minimal onsite consequences to detectable offsite consequences. In developing the representative set of potential accidents, the NRC staff considered past experience at fuel cycle facilities handling UF_6, experience at Urenco facilities in Europe, and the CEC design described in the SAR. The only accident deemed to be possible at the CEC which could have significant offsite consequences is release of a large quantity of UF_6 from a cylinder containing liquid UF_6. Under normal operating conditions, UF_6 is present in the liquid state only in cylinders inside of autoclaves. Attainment of the elevated temperature and pressure required to breach a cylinder and autoclave is prevented by heater shutoff circuits activated by occurrence of high temperature or pressure. These protection circuits are provided for each autoclave. The protection circuits are designated important to safety and are designed to maintain function in the occurrence of design basis events. The severity of the design basis events is specified in the ANPR. Uncontrolled fire is an additional source of energy which could possibly lead to rupture of a UF_6 cylinder. The CEC design precludes occurrence of this scenario through absence of combustible material in areas where significant quantities of UF_6 are present and limitation on fuel inventories on UF_6 cylinder transporters. Due to the design features and the use of redundant, diverse protection circuits, the NRC staff concludes that occurrence of potential accidents at the CEC does not pose an undue risk to public health and safety.

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Four general and seven limited partners plan to construct, operate, and decommission the CEC. The applicant estimates that hard construction costs will be \$816 million in 1992 dollars. Approximately 30 percent equity financing will be used. The NRC staff reviewed the financial status of each of the partners, including reports of shareholder equity, cash flow, and cash on hand to evaluate the source of and reliability of funds. The applicant commits to maintaining \$200 million in nuclear liability insurance. The NRC staff reviewed the applicant's estimate of product market prices and considered project risk. On the basis of this review, the NRC staff concludes that the financial risk of the project should not affect the protection of public health and safety.

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Facilities licensed by the NRC under 10 CFR Parts 40 and 70 are required to be decommissioned by the licensee in order to permit release of the site and facilities for unrestricted use and to terminate the license. In order to decontaminate and decommission (D&D) the facility, the applicant proposes to incorporate specific features into the design which will facilitate D&D, characterize the facility and site after termination of enrichment operations, prepare a detailed decommissioning plan, complete D&D activities, and complete a final site survey. The entire D&D process is estimated to require 7 years and will involve installation of two new facilities for decontamination. The applicant's estimate of D&D cost is approximately $518 million dollars, including $485 million estimated for the cost of dispositioning of depleted UF_6 tails. The NRC staff concludes that the applicant's proposed procedures and funding are adequate to D&D the facility. In addition, the applicant has committed by license condition to updating the decommissioning funding plan at least every 5 years, starting from the time of issuance of the license.

2 SITE CHARACTERISTICS

This chapter describes and analyses those site characteristics which may affect operations or accidents and reviews the applicant's analysis which postulates the severity of natural phenomena included in the plant design basis. These site characteristics--demographic, hydrologic, meteorological, and seismological factors--determine the spatial and temporal distribution of impacts of releases from a facility during normal operation or accidents. This chapter also presents NRC staff evaluations of the proposed design basis natural phenomena and of other site characteristics.

This chapter consists of five sections. The first section describes the geography and population distribution around the site, with emphasis on the nearby residents most likely to be affected by accidental releases. The second section describes nearby industrial, transportation, and military facilities, the operation of which could be affected by accidents. The third section describes the meteorological conditions, which play a role in atmospheric transport from the facility to potential receptors, and reviews the applicant's analysis of the design basis high wind and tornado. The fourth section describes the surface water flow system which could transport releases from the facility and reviews the applicant's analysis of the design basis flood. The fifth section describes the geologic setting of the site and reviews the applicant's analysis of the design basis earthquake. The data and descriptions presented are drawn from the Louisiana Energy Services' (LES) Claiborne Enrichment Center (CEC) Safety Analysis Report (SAR) (LES, 1993a) and Environmental Report (ER) (LES, 1993b).

2.1 Geography and Demography

2.1.1 Site Location

The site for the proposed CEC is located in Section 3, T21N, R6W, of Claiborne Parish, northwest Louisiana, approximately 8 kilometers (5 miles) northeast of the town of Homer, LA, and 80 kilometers (50 miles) east-northeast of Shreveport, LA. The location of the CEC site within the region is shown in Figure 2.1; its location within the local area is shown in Figure 2.2. The site covers 179 hectares (440 acres); the controlled area, situated at the center of the site, covers 28 hectares (70 acres) and includes seven main buildings enclosed by a fence.

2.1.2 Site Description

Topography in the area around the CEC property is characterized by rolling hills, with ground-level elevations ranging from 60 to 100 meters (200 to 330 feet) above mean sea level (MSL). Elevations within the CEC property range from 85 to 104 meters (280 to 340 feet) MSL, for the southern and central portions, respectively. Generally, the terrain can be described as ranging between flat and gently rolling.

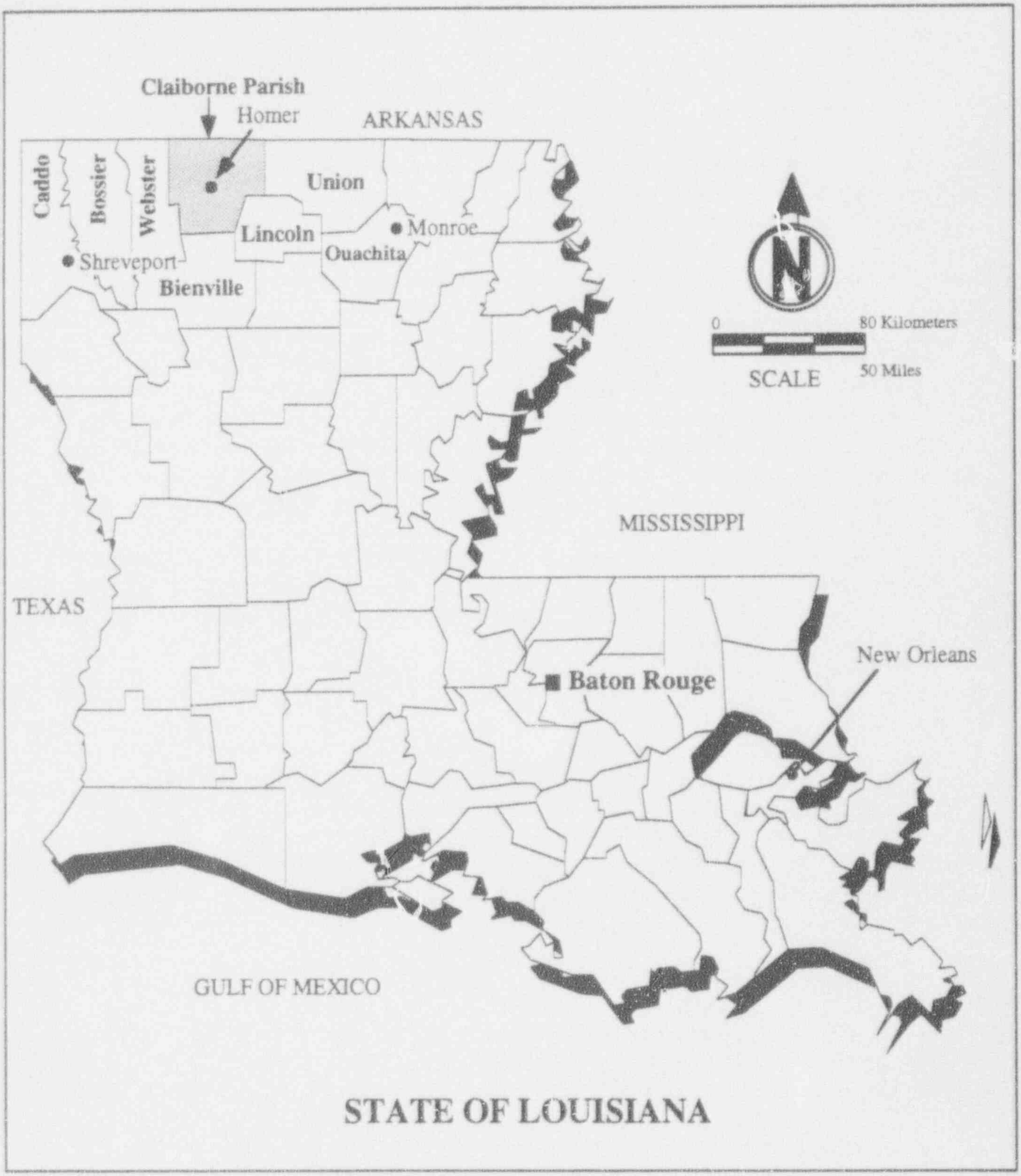

Figure 2.1 Regional map of the CEC area

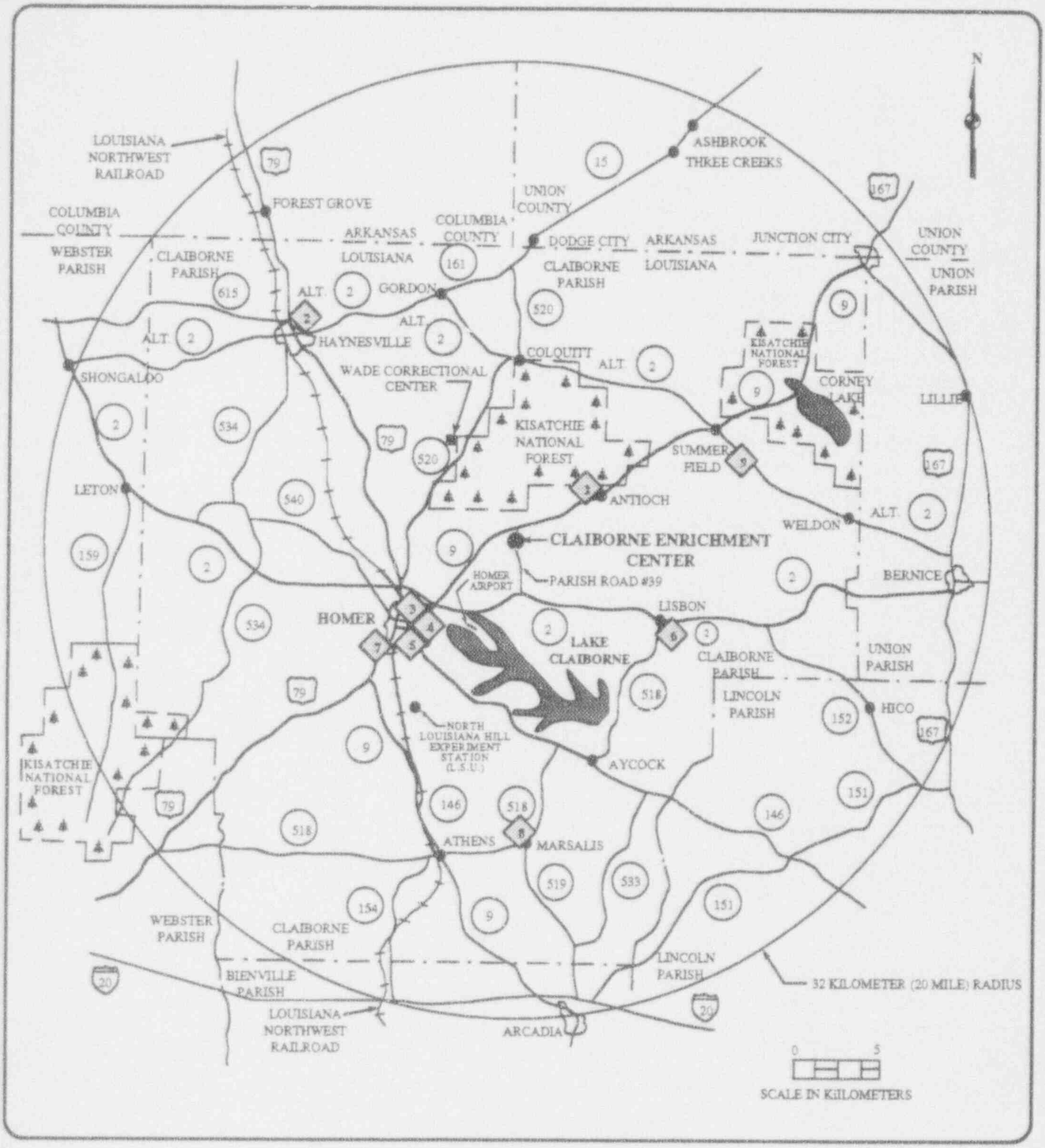

Figure 2.2 Local map of the CEC area

The entire site is part of the Ouachita River drainage basin. Drainage for the western and southern portions of the site is provided by small unnamed creeks. These creeks join in the southwestern portion of the site and flow into Bluegill Pond, at the head of Cypress Creek, which flows into Lake Claiborne. For the eastern portion of the property, drainage is directed into Lake Avalyn, a small lake at the head of an unnamed stream, which flows into McCasland Creek. A more detailed description of site drainage is presented in Section 2.4.

Within the controlled area, a plant drainage system routes normal surface runoff into catch basins, a hold-up basin, and Bluegill Pond. Flooding of the developed area is prevented by the grading of the plant yard, which routes the run-off away from the structures.

Vegetation in the vicinity of the site is thick and composed mostly of pine, with some oak in the bottom land and moderate to dense underbrush. Extensive deforestation occurred between the late winter of 1989 and the early summer of 1990. There are no plans for reforestation of the clear-cut areas; instead, they will be allowed to reforest naturally.

On-site access roads could serve for movement of UF_6 cylinders or centrifuge components. Incoming vehicles would enter the south gate at the security station, then drive either to the Cylinder Receipt and Dispatch Building (CRDB) or to the Centrifuge Assembly Building (CAB). In addition, there are paved roads between the CRDB and the Separations Building, and within the storage areas, for the use of facility vehicles. Figure 2.3 shows the plan of the facility and the configuration of the on-site access roads.

2.1.3 Boundaries

All activities within the fenced, 28-hectare (70 acre), controlled area are related to plant operation. LES has the authority to determine all activities within this area. Any possible future activities not related to the operation of the plant within this area will not interfere with operations.

2.1.4 Population Distribution

2.1.4.1 Population in the Vicinity of CEC

The population of Claiborne Parish has been stable, though slightly increasing, over the last three decades. In 1970, the population was 17,024; in 1980, 17,095; in 1990, 17,405 (U.S. Department of Commerce, 1992a). Most of the population in the parish is clustered along Parish Roads 39 and 9, in the towns of Homer and Haynesville, and in the Wade Correctional Center. Both Claiborne Parish and the area around the CEC are sparsely populated. The population density for the parish was 8.9 persons per square kilometer (23.1 per mi^2) in 1990 (U.S. Department of Commerce, 1992a).

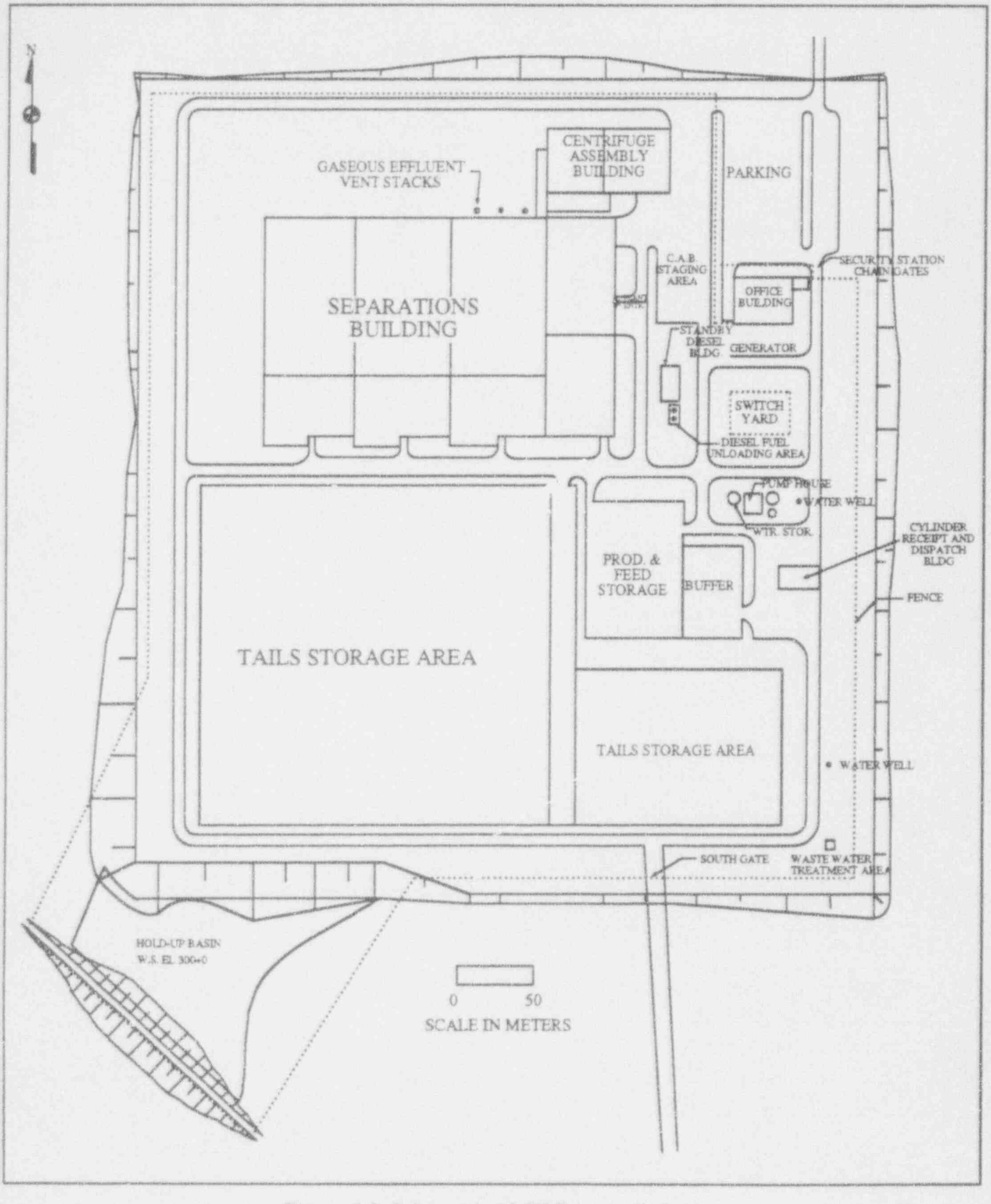

Figure 2.3 Schematic of CEC controlled area

A large transient population clusters mainly around Lake Claiborne and, to a lesser extent, other recreational facilities such as Corney Lake and the Kisatchie National Forest. Most of these visitors engage in boating, fishing, hunting, and skiing. The transient population at Lake Claiborne State Park alone was 48,200 in fiscal 1988-89. The permanent residence nearest to the CEC is located approximately 475 meters (1560 feet) north-northeast of the plant stacks.

2.1.4.2 Population within 80 Kilometers of CEC

The closest major urban center is Shreveport, LA, which is 80 kilometers (50 mi) southwest of the facility and has a population of 205,000. El Dorado, AR, lies about 80 kilometers northeast from the facility and has 25,000 residents (U.S. Department of Commerce, 1992a). Total population of the adjacent parishes is 472,000. Total population within an 80-kilometer radius of the site, including as many as 1,167 inmates of the Wade Correctional Center, is 349,000. The distribution of population by direction and distance is summarized in Table 2.1.

Table 2.1 Population distribution within 80 kilometers of the CEC

Direction	Distance (km)									
	0.8	2.4	4.0	5.6	7.2	12.0	24.0	40.0	56.0	72.0
S	6	32	49	15	244	196	2361	2524	1315	2157
SSW	0	3	23	29	289	288	1404	1981	2646	4531
SW	0	0	9	3	213	1727	1370	18675	6539	5213
WSW	0	3	40	0	51	2055	752	2178	7308	74494
W	0	9	3	0	6	481	877	3273	1915	4935
WNW	12	9	3	3	20	603	1494	5830	6426	2424
NW	9	3	9	0	0	47	3020	2348	3471	5201
NNW	6	0	0	29	1170	227	402	3152	14291	2944
N	20	0	0	0	0	218	407	833	1977	2435
NNE	17	20	17	17	0	182	540	2621	10373	1906
NE	9	12	6	12	0	273	1162	5894	20851	1220
ENE	3	6	32	26	6	253	895	1839	1704	3333
E	0	0	0	9	17	126	2367	1430	5238	3501
ESE	0	6	23	6	0	64	1240	1974	3005	8772
SE	0	23	0	15	29	45	2047	23814	3352	2940
SSE	0	26	0	3	34	148	1531	7676	3162	12419

2.2 Nearby Industrial, Transportation, and Military Facilities

Nearby industrial, transportation, and military facilities could affect or be adversely affected by CEC operations. This section briefly describes the characteristics of these facilities and supports the NRC staff finding on potential interactions. Because of the distances between the CEC and all other industrial, transportation, and military facilities, the NRC staff concludes that the CEC would not be significantly affected by current or planned activities, and none of these activities would be adversely affected by CEC operations.

2.2.1 Industrial Facilities

Industrial facilities in the vicinity which could affect CEC operation include oil and gas production and miscellaneous manufacturing operations. This section summarizes the number and types of these facilities.

2.2.1.1 Oil and Gas

Claiborne Parish has 1,176 producing oil wells and 202 producing gas wells (LP&L, 1990). There are 31 wells within 8 kilometers (5 miles) of the CEC facility. Of these, 19 were producing oil, 10 were producing gas, and 2 were injection wells in 1990 (LES, 1993b). The nearest producing well is a gas well approximately one kilometer (0.6 miles) southeast of the CEC.

The applicant has analyzed the potential risk posed by oil and gas production in the vicinity of the site (LES, 1993a). The applicant selected an oil- and gas-producing well located near the southwest corner of the site as the basis for analysis. This well is the producing well nearest the CEC. The analysis demonstrated that due to the nature of the local topography, oil spilt from the well would flow away from the site and not pose a potential safety threat. Analysis of the possibility of explosion of gas released from the well was based on development of an estimate of the maximum quantity of gas which would be released in a rupture. Because the well piping includes a valve designed to close on over- or under-pressure signals and the local metering station includes backflow prevention, the maximum estimate gas release is determined by the volume of piping between the valve and transmission pipeline. The quantity of gas which would be released from the well would be less than that required to initiate a vapor-phase explosion.

The NRC staff reviewed the applicant's analysis and finds that it is reasonable and supports the conclusion that operation of off-site oil and gas wells does not pose a risk to operation of the CEC.

2.2.1.2 Manufacturing

In 1990, Claiborne Parish had nine manufacturing operations employing a total of 684 residents in petroleum products, timber/wood products, plastics, garments, and packaging (LP&L, 1990). Table 2.2 lists these companies, their primary business, and number of employees. Given the nature and separation of the facilities, the NRC staff concludes that no significant interactions with the CEC would occur.

Table 2.2 Manufacturers in Claiborne Parish (1990)

COMPANY	PRIMARY BUSINESS	EMPLOYEES
Emont	Industrial gloves	300
Ludlow Corp.	Packaging products (industrial)	200
Claiborne Gasoline Co.	Petroleum products	46
Harmon Wood Co., Inc.	Pulpwood, longwood, logs, wood, chips	30
Woodsmith, Inc.	Molding, hardwood	43
Beacon Plastics, Inc.	Plastic injection molding (custom)	20
Delat Draperies	Draperies, bedspreads	15
Industrial Packaging	Packaging products (industrial)	15
Laark Fashions	Women's custom clothing	15

Source: LP&L, Claiborne Parish Profile, 1990, p.22.

2.2.1.3 Nuclear

Because no nuclear facilities exist within 32 kilometers (20 miles) of the CEC site, any potential environmental problem at the CEC would be an isolated occurrence and, in any emergency, would thereby preclude interactions with other nuclear facilities. The nearest facility is the Grand Gulf Nuclear Power Station near Port Gibson, MS, approximately 215 kilometers (135 miles) southeast of the CEC. More distant nuclear facilities include Arkansas Nuclear One in northern Arkansas (about 265 kilometers [165 miles] north), River Bend in southern Louisiana (about 290 kilometers [180 miles] south), Waterford 3 in southeast Louisiana (about 400 kilometers [250 miles] away), and Comanche Peak in Texas (about 430 kilometers [270 miles] west) (NERC, 1988). Given the great distances between facilities, the NRC staff concludes that no interactions with the CEC would occur and that doses in the CEC area from the other nuclear facilities would be insignificant.

2.2.2 Transportation

This section discusses air and surface transportation by roads, railroads, airports, and waterways, and land links by pipelines and power lines. Except for Parish Road 39, there would be no changes in traffic patterns other than a moderate increase in traffic along Roads 2 and 9.

2.2.2.1 Roads

State Roads 9 and 154 and Federal Highway 79 link Homer with Interstate 20 south of the parish. State Road 2 links Bernice, Lisbon, Homer, and Cotton Valley; State Road 146 links Homer and Ruston; State Road 9 links El Dorado, AR; Athens and Homer, LA; and Interstate 20. Federal Highway 79 links Magnolia, AR, to Haynesville and Homer, where it veers southwest towards Minden. The regional and local access roads to the CEC are shown in Figure 2.2.

2.2.2.2 Railroads

The nearest major railroad is the Louisiana Northwest Railroad, which connects with the Southern Pacific Lines at its northern terminal, runs south parallel to Federal Highway 79 from the Arkansas border to Homer, then heads south along State Roads 9 and 154 to link the parish to the Mid-South Railroad, which runs parallel to and south of Interstate 20. Two freight trains serve customers and deliver to connections 5 days a week, with one-stop train service 2 days a week. System capacity is 285,705,000 gross kilograms (628 million pounds) per train (Ralston, R., Louisiana Northwest Railroad, Personal Communication, May 22, 1991).

2.2.2.3 Airports

The parish has two general aviation airports, both of which operate during daylight hours, 7 days a week. One is the Homer Municipal Airport, which lies 5 kilometers (3 miles) east of the city and has a lighted 1,000-meter (3280-foot) runway. Airport traffic averages four aircraft per day. The second is Haynesville Airport, which has a lighted 900-meter (2950-foot) runway. The nearest commercial airports outside the parish are located in El Dorado, AR, fewer than 65 kilometers (40 miles) away, and in Shreveport, 80 kilometers (50 miles) away (LP&L, 1990, and Homer Chamber of Commerce, 1990). Figure 2.2 shows the location of the Homer Municipal Airport vis-a-vis the CEC.

Air traffic within the parish is not expected to increase significantly because of CEC operation. No current or foreseen population growth would increase local air traffic over the projected life of the facility. The applicant analyzed the probability of an airplane crashing into the site (LES, 1993a). The analysis considered airplane types, flight frequencies and trajectories, and plant area in order to develop an estimate of 1.3×10^{-9}, or 1.3 in a billion, crashes per year.

The NRC staff reviewed the airplane crash analysis and concludes that it is reasonable and consistent with recommended methods (NRC, 1987a). On the basis of this low probability of occurrence, the NRC staff concludes that the combined operation of air transportation and the CEC does not pose an unreasonable risk to public health and safety.

2.2.2.4 Waterways

Claiborne Parish has no waterways. The two nearest points with access to waterways are the Red River port of Shreveport, which lies 95 kilometers (60 miles) away, and the Ouachita River, which lies 110 kilometers (70 miles) away. Both waterways are used for river transportation (LP&L, 1990).

2.2.2.5 Oil and Gas Pipelines and Electric Power Lines

No gas or oil pipelines cross the CEC site. Five pipelines are located within 8 kilometers (5 miles) of the CEC site (approximate shortest distance to the CEC facility indicated), as shown in Figure 2.4:

- Associated Gas -- An active, 5-centimeter (2-inch) pipeline running southeast-northwest, then turning and heading north, 7.2 kilometers (4.5 miles) west of the facility

- Associated Gas -- An active, 20-centimeter (8-inch) gas pipeline running northwest-southeast, 3.8 kilometers (2.4 miles) northeast of the facility

- Citgo Crude -- An active, 15-centimeter (6-inch) oil pipeline running west-northeast, approximately 1.6 kilometers (1 mile) north of the facility

- Tet Products -- An idle, 20-centimeter (8-inch) pipeline running southwest-northeast, 3 kilometers (2 miles) southeast of the facility

- UG Gas -- An active, 50-centimeter (20-inch) pipeline running east-west, 6.0 kilometers (4 miles) south of the facility.

The applicant has analyzed the potential hazard of operation of nearby oil and gas lines (LES, 1993a). The analysis considered the 50-cm (20-inch) United Gas line located 6 kilometers (4 miles) south of the facility and referred to prior analysis completed in relation to operation of the Susquehana Steam Electric Station. This analysis evaluated the operation of a 105-centimeter (40-inch) pipeline located 2 kilometers (6700 feet) from the power plant. Evaluation of failure of the pipeline considered blast over-pressure, thermal radiation, and missiles and concluded that plant operation would not be affected by the pipeline failure. Because the pipeline under evaluation has lower capacity and lower operating pressure, the applicant concluded that pipeline failure would not affect CEC operation.

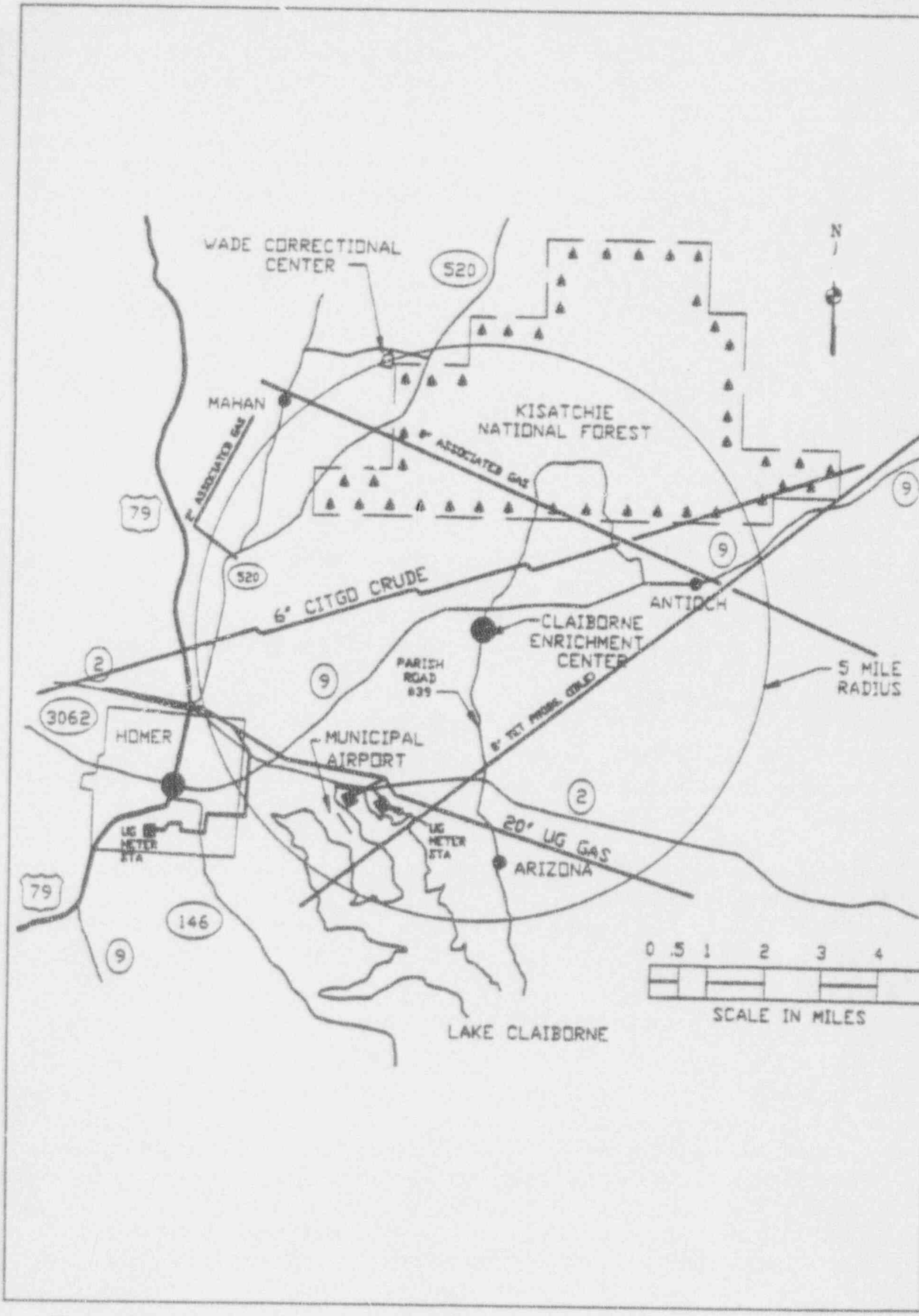

Figure 2.4 Location of oil and gas pipelines within 8 kilometers of the CEC site

The NRC staff reviewed the applicant's analysis of gas pipeline operation and concludes that operation of such lines at the present distances from the CEC does not pose a risk to safe operation of the facility.

Electricity in the parish is provided by Claiborne Electric Power Cooperative, Inc., and Louisiana Power & Light Company (LP&L). Claiborne Electric is part of Louisiana's Electric Cooperative System and is headquartered in Homer (LP&L, 1990). LP&L operates an interconnected integrated electric system covering a 50,000 square-kilometer (20,000 square-mile) area in 46 of the 64 parishes of Louisiana. The nearest LP&L transmission line goes through a substation in Haynesville, northwest of the CEC site; another line goes through a substation in Bernice, east of the CEC site. A 115-kilovolt overhead line from each of these substations will be routed to a switchgear on the CEC site. Each line will be capable of meeting all facility requirements. Only LP&L will provide power for the CEC (LES, 1993a).

2.2.3 Military Facilities

There is no military presence in Claiborne Parish. Moreover, there are no military bases, bombing ranges, munitions plants, or missile installations within 8 kilometers (5 miles) of the facility. The nearest military facility is the Barksdale Air Force Base in Bossier City, 60 kilometers (40 miles) west of Homer (Brakefield, M., Homer Chamber of Commerce, Personal Communication, October 28, 1992 and Rand McNally, 1991).

2.3 Climatology and Meteorology

This section describes the climatological and meteorological conditions at the site, reviews the applicant's analysis of the design basis high wind and tornado, and summarizes the results of atmospheric dispersion analysis appropriate for normal operation and accidents.

Climatological and meteorological conditions can initiate or increase accidental releases of radioactive or other hazardous material and would disperse such material released during normal operation and accidents. Radioactive material would be released to the atmosphere during normal operation of the CEC facility and could also be released to the atmosphere during accidents.

2.3.1 Climatology

The climate of north-central Louisiana is transitional between the subtropical, humid climate of the Gulf of Mexico and the continental climates of the great plains and middle west. The average annual temperature is 18.6 °C (65.4 °F). The rural terrain of gently rolling hills allows unobstructed air flow from any direction. Summer months are quite warm and humid, with afternoon temperatures above 30 °C (85 °F) and afternoon humidity in the 60- to 75-percent range. Annual rainfall, which totals over 127 centimeters (50 inches) (only October averages fewer than 3 inches), occurs primarily in moderate to heavy rains usually associated with thunderstorms, especially in spring and summer. During winter, masses of moderately cold air periodically move through the area. Snowfall and prolonged cold spells are unusual, and measurable snow during a year is rare. Ice storms and freezing rains often damage power lines and make traveling hazardous. Limited climatological data for the site are gathered in Homer; more complete data are gathered at Shreveport, which has the closest National Weather Service Station. The CEC Draft Environmental Impact Statement (DEIS) (NRC, 1993a) provides additional information.

2.3.1.1 Winds, Tornadoes, and Storms

Winds and Tornadoes

In Claiborne Parish, high winds are most frequently associated with thunderstorms, far less frequently with hurricanes, the winds of which may be sustained but are rarely destructive. Tornadoes are not common because of the rolling hills and forest cover around the site, and the abundant water in rivers and lakes which moderates temperatures.

The applicant has provided an assessment of straight and tornado wind speed probabilities (McDonald-Mehta Engineers, 1990) to characterize the wind hazard for the CEC site. The analysis of straight wind speed was derived from maximum annual wind speed data from Shreveport and Barksdale Air Force Base. Meteorological tower heights at both locations are about 6 meters (20 feet) as opposed to the standard height of 10 meters (33 feet). Measurements at this non-standard tower height result in lower reported wind speeds because of friction with the ground, vegetation, and structures. Data were corrected to the standard tower height and converted to fastest-mile wind speed.

Hurricanes dissipate too much over southern Louisiana to pose a severe wind damage threat to the site in the northern part of the state. Their winds are usually not destructive, but severe flooding can be expected near, but not at, the site. Hurricane Andrew, which struck Louisiana on August 26, 1992, is a good example. As reported by the National Hurricane Center, winds from this Force-3 hurricane neared 60 m/s (115 knots) at landfall. About 3 1/2 hours later, winds had fallen to 40 m/s (80 knots); after 9 1/2 hours, with winds at 25 m/s (50 knots), Andrew was downgraded to a tropical storm; after 21 1/2 hours, with winds at 15 m/s (30 knots), Andrew was downgraded to a tropical depression (NHC, 1992). The distance traveled by Andrew by the time it was downgraded to a tropical depression approximates the distance that would be covered if a hurricane went straight to Homer. Thus, a wind speed of 15 m/s (30 knots) would be the sustained wind speed expected from a Force-3 hurricane. More powerful hurricanes would be expected to result in windspeeds only somewhat higher because most of the force of the storms would be spent near the coast, and additional energy would not be available to them over land.

Applicant analysis of tornado probabilities followed the method of NUREG/CR-3058 (McDonald, 1983). The method required the development of tornado area-intensity and occurrence-intensity relationships. The area-intensity relationship is a function of mean area of the damage path and the wind velocity, and was developed on the basis of National Severe Storm Forecast Center data for the five-degree square represented in Figure 2.5. The occurrence-area relationship was developed from data on 632 tornadoes reported from 1950 to 1987 for the three-degree square represented in Figure 2.5. Average occurrence is equivalent to 16.6 tornadoes per year in an area the size of Indiana. There is no significant statistical difference in tornado frequency between these slightly differently defined areas. Wind speed, damage area, and occurrence data for the differing classes of tornadoes are summarized in Table 2.3.

The fastest-mile wind-speed probabilities were combined with the tornado data to obtain an overall wind-speed probability data set. The annual probabilities of exceeding a given wind speed are summarized in Table 2.4. The Advance Notice of Proposed Rulemaking (ANPR) (NRC, 1988a) specified a $1x10^{-4}$/yr frequency of occurrence as the design basis for enrichment facilities. The atmospheric pressure change and rate of atmospheric pressure change estimated by the applicant for the design basis tornado are 1915 pascals (Pa) (40 psf) and 958 Pa/s (20 psf/s), respectively. The representative missiles were a 5x10-centimeter (2x4-inch) timber traveling with a horizontal speed of 45 m/s (100 mph) and a 7.5-centimeter (3-inch) diameter steel pipe traveling with a horizontal speed of 22 m/s (50 mph).

The NRC staff reviewed the applicant analysis, compared the results with NUREG/CR-3058 and ANPR guidance, and concludes that the results provide an acceptable design basis for the CEC.

2.3.1.2 Storms

The most common storms are thunderstorms, which occur during all months. Shreveport data shows that an average of 55.9 thunderstorms occur each year, with the summer months having a higher frequency (see Table 2.5). Hurricanes may have heavy rains associated with them (Andrew dumped up to 25 centimeters [10 inches] on a near-coastal Mississippi town), but they move fairly rapidly, and inland areas rarely get more than a several centimeters. The heaviest rains are associated with very slow-moving tropical depressions.

Severe snow or ice storms, with accumulations of 2.5 centimeters (1.0 inch) or more, are infrequent and occur from November to March. There is a 60-percent chance of such a storm in any given year, a 30-percent chance in January alone. Although such storms are infrequent and the snow or ice rarely remains more than a few days, the impacts are much more severe in this area than in most of the continental U.S. because of the lack of equipment for and experience with such conditions.

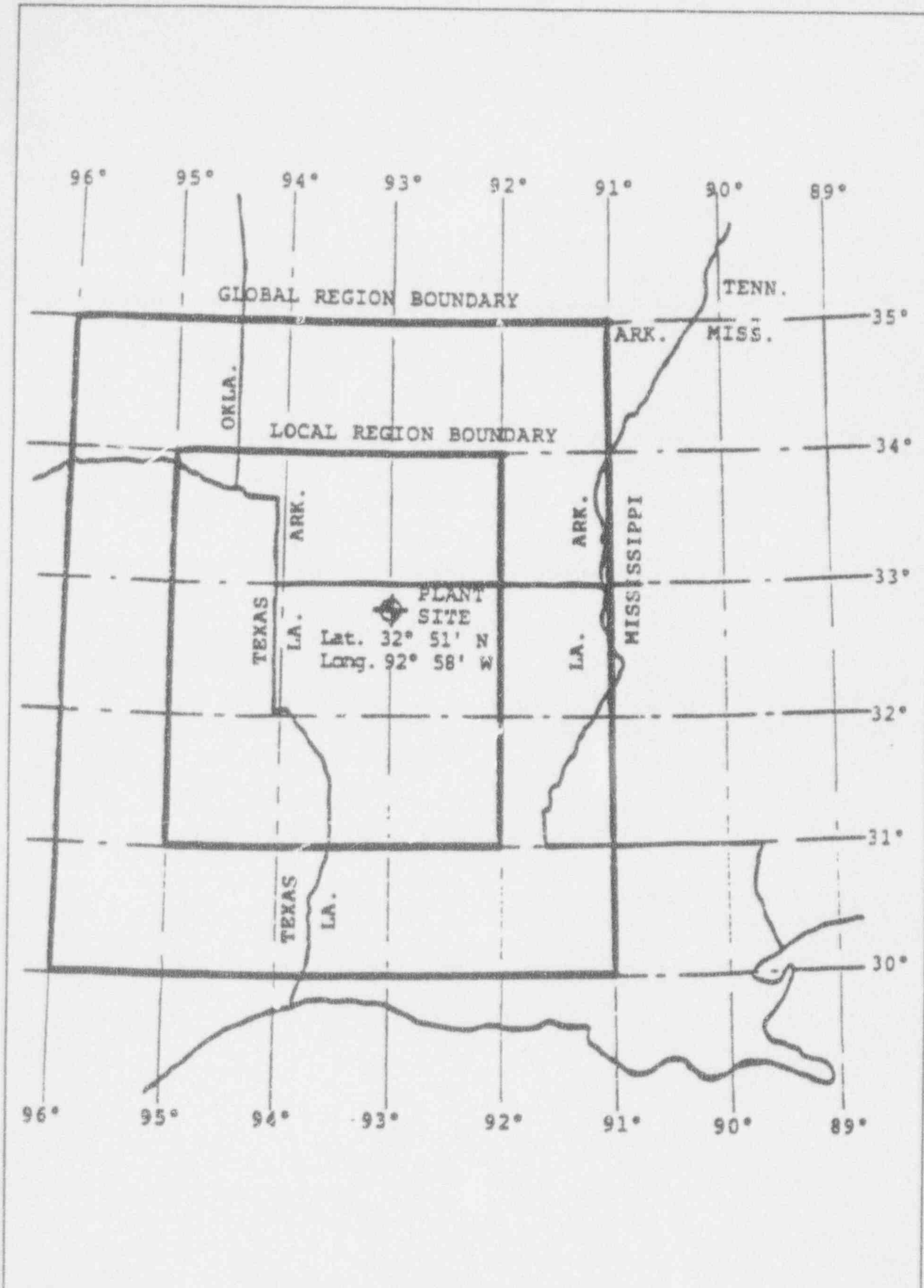

Figure 2.5 Area evaluated in tornado study

Table 2.3 Frequency of damaging tornadoes

F-SCALE	MEDIAN WIND SPEED (m/s)	MEAN AREA DAMAGED[a] (hectares)	NUMBER OF TORNADOES[b]
F0 (18-32 m/s) Light damage: antennas and chimneys lightly damaged; branches break off trees; old trees with hollow trunks break and fall.	25	4.7	108
F1 (33-50 m/s) Moderate damage: roofing peeled off; windows broken; trailer homes displaced or overturned; trees on soft ground pushed over; some trees snapped.	41.3	56	256
F2 (51-70 m/s) Considerable damage: roofs torn off frame houses but strong walls left upright; weak structures and outbuildings destroyed; trailer homes demolished; cars blown off roads.	60.3	217	184
F3 (71-92 m/s) Severe damage: some rural buildings demolished; roofs and some walls torn off well-constructed buildings; trains overturned; cars lifted off the ground and rolled; most trees uprooted or snapped; block structures often leveled.	81.4	1316	74
F4 (93-116 m/s) Devastating damage: well-constructed houses leveled; structures with weak foundations lifted, torn, and blown some distance; trees debarked by flying debris; gravel and sand fly in wind; cars blown, rolled, and destroyed; large missiles generated.	104.4	533	9
F5 (117-142 m/s) Incredible damage: strong frame houses lifted, carried long distances, and disintegrated; steel-reinforced concrete structures badly damaged; automobiles lifted and carried more than 100 yards.	129.4	2535	1

[a] The mean area of tornado damage in the 1950-1987 period between 30° and 35°N latitude and 91° and 96° W longitude

[b] The number of reported tornadoes in the 1950-1987 period between 31° and 34°N latitude and 9[illegible] and 95° W longitude

Table 2.4 Fastest-mile wind-speed probabilities

TYPE OF WIND	WIND SPEED[a] (m/s)	ANNUAL PROBABILITY
Straight	22.8	1:10
Straight	27.4	1:100
Straight	31.9	1:1,000
Straight, Tornado	34.4	1:3,000
Torr.ado	51.4	1:10,000
Tornado	78.2	1:100,000
Tornado	99.7	1:1,000,000
Tornado	122.5	1:10,000,000

Source: McDonald-Mehta Engineers, 1990
[a] Wind speed represents a 2-second gust

Table 2.5 Shreveport storm summary

MONTH	PREVAILING WIND DIRECTION[a]	DIRECTION (DEGREES)	MEAN SPEED (m/s)	MAX 1-MIN SPEED (m/s)	PEAK GUST SPEED (m/s)	THUNDERSTORMS	SNOW OR ICE OF 2.5 CM OR MORE
JAN	S	220	4.2	17	18	1.8	.3
FEB	S	270	4.3	18	22	2.7	.2
MAR	S	290	4.6	18	26	4.9	.1
APR	S	280	4.4	23	28	5.6	0.0
MAY	S	280	3.8	17	23	7.1	0
JUN	S	160	3.4	17	25	7.2	0
JUL	S	290	3.2	21	30	8.0	0
AUG	S	250	3.1	17	22	6.6	0
SEP	ENE	190	3.3	20	17	4.0	0
OCT	SSE	310	3.3	16	18	2.7	0
NOV	S	290	3.8	17	21	3.0	0.[b]
DEC	S	140	4.0	17	29	2.2	0.1
YEAR	S	280	3.8	23	30	55.9	[illegible]

Source: NOAA, 1990(a)
[a] Through 1963
[b] The value is between 0.1 and 0.05

2.3.2 Meteorology

The meteorological factors affecting dispersion and potential exposure of receptors include wind speed, atmospheric stability, and mixing heights. Although no meteorological stations are in the immediate vicinity of the site, three reasonably close sites provide suitable meteorological data. These sites are Shreveport, LA; Monroe, LA; and El Dorado, AR, located 72 kilometers (45 miles) west-southwest, 92 kilometers (57 miles) east-southeast, and 56 kilometers (35 miles) northeast of the CEC site, respectively. National Weather Service (NWS) data shows that the regional temperatures and precipitation increase from north to south and from west to east. The differences among the data from these stations and the Homer station are trivial and imply that the climatology in the region is nearly uniform. A more complete description of regional and local meteorological conditions, including the joint distribution of wind conditions, is presented in the NRC's CEC DEIS (NRC, 1993a).

The frequencies at which winds blow in a particular direction and at various speeds are usually represented graphically by a "wind rose." The wind roses for all three sites are very similar, but the data from Shreveport and Monroe, which have virtually identical stability classifications, show more stability than data from El Dorado. Both Shreveport and the CEC site are located on high hills, and no intervening geologic features or large bodies of water can differently affect the weather between the two sites. El Dorado is located in a valley typical of southern Arkansas, and valley temperatures are often lower than those of surrounding areas because of shading during the day and radiational cooling at night. These differences create inversion layers differing from the predominant mixing layer. Because of significant geophysical and meteorological differences between El Dorado and the CEC site, El Dorado data should be excluded from consideration. Because of the geophysical and meteorological similarities between Shreveport and the CEC site, the meteorological similarities between Shreveport and Monroe, and the proximity of Shreveport to the CEC site, the NRC staff considers the Shreveport data the most appropriate for dispersion modeling of the CEC site (Ethridge, 1991). The wind rose for average annual conditions at Shreveport is presented in Figure 2.6.

Onshore airflow from the Gulf of Mexico causes southerly winds to prevail most of the year. Cold fronts cause northerly flows. The frequency distribution of wind speeds and stability classes for Shreveport is presented in Table 2.6. The annual average wind speed at the 10-meter level is 3.4 m/s (7.6 mph); calms are reported 12.9 percent of the time. Neutral stability (Pasquill type D) conditions predominate, occurring 40 percent of the time at the site. Moderately stable (Pasquill type F) and extremely stable (Pasquill type G) conditions occur 15 and 10 percent of the time, respectively.

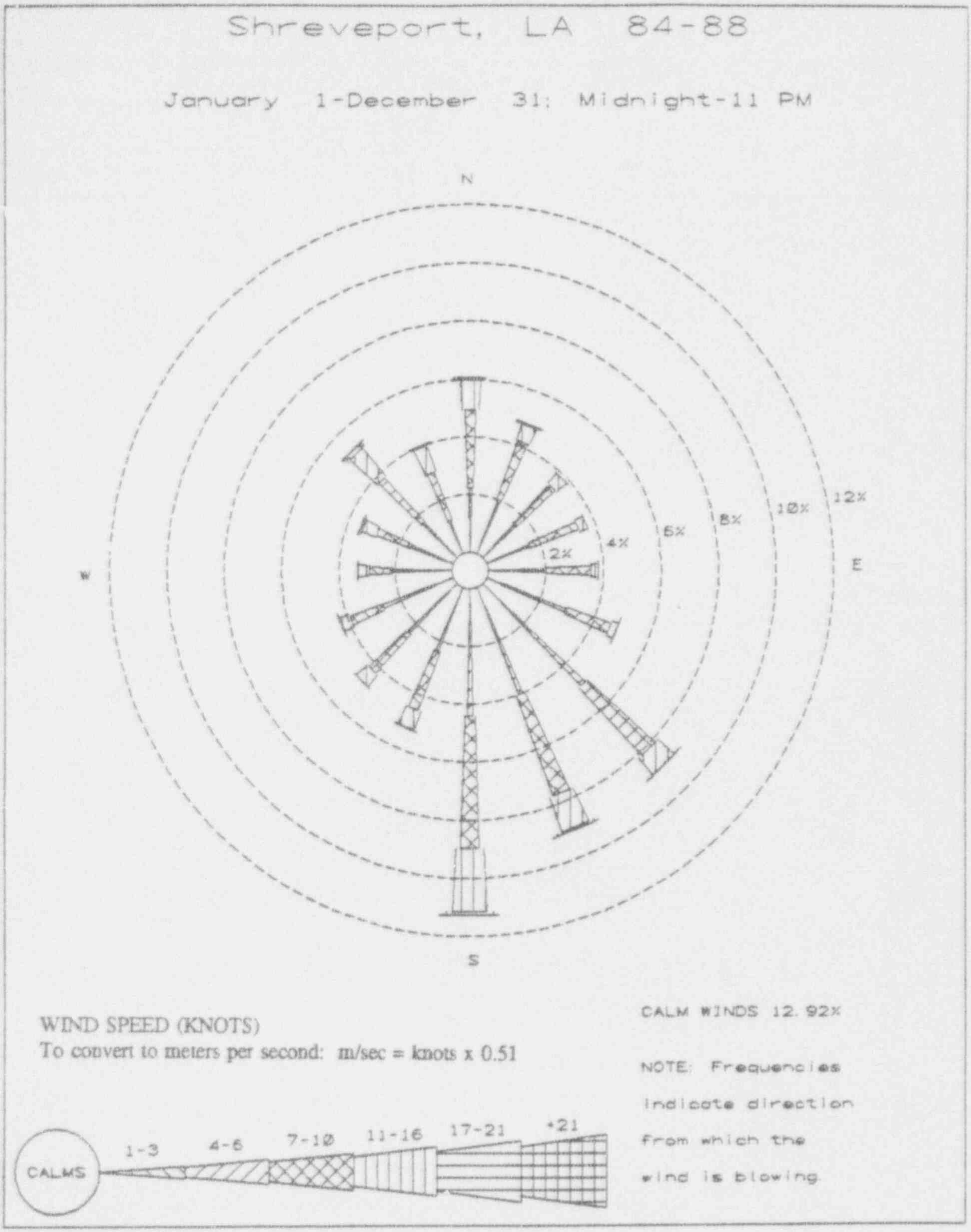

Figure 2.6 Wind rose for Shreveport, LA (1984-1988) (NOAA, 1990)

Table 2.6 Percent of time winds blow at each wind speed

STABILITY CLASS	WIND SPEED (m/sec)						
	1.5	2.5	4.3	6.8	9.5	12.5	TOTAL
A	0.6	0.9	0.0	0.0	0.0	0.0	1.5
B	1.7	3.6	2.6	0.0	0.0	0.0	7.9
C	0.7	3.5	7.7	1.0	0.0	0.0	12.9
D	1.7	8.5	17.8	10.9	0.9	0.1	39.8
E	0.0	6.4	6.3	0.0	0.0	0.0	12.7
F	4.1	10.6	0.0	0.0	0.0	0.0	25.0
TOTAL	19.0	33.5	34.4	11.9	0.9	0.1	99.8

Mixing heights and controlling inversion layers typically have large diurnal variations. Seasonal morning and afternoon mixing heights and wind speeds for the site are shown in Table 2.7 (Holzworth, 1972). Summer afternoons have the highest mixing heights and lowest wind speeds for afternoons, and spring mornings have the highest mixing heights and highest wind speeds for mornings. The mixing heights and wind speeds are fairly typical of the interior United States.

Table 2.7 Seasonal mixing heights and wind speeds

PERIOD	MORNING		AFTERNOON	
	MIXING HEIGHT (m)	WIND SPEED (m/sec)	MIXING HEIGHT (m)	WIND SPEED (m/sec)
WINTER	500	6.0	1,000	6.5
SPRING	550	6.0	1,400	7.0
SUMMER	500	4.0	1,800	5.0
FALL	400	4.5	1,400	5.5
ANNUAL	500	5.0	1,400	6.0

Source: Holzworth, 1972

Pollution episode conditions defined by the National Air Pollution Potential Forecasting Program are the combination of mixing heights under 1,500 meters (4,920 feet), wind speeds under 4.0 m/s (8.9 mph), and no significant precipitation. Holzworth's maps show that Shreveport, the closest station for which there are upper-air data, reported 13 episodes lasting at least 48 hours and 32 episode-days during the 5-year period of 1960-1964. Fall episodes predominated, with no episode lasting more than 4 days. It appears that spring afternoons may offer the best dispersion combination of mixing heights and wind speeds and fall mornings the worst.

2.3.2.1 Onsite Meteorological Measurements

Installation of meteorological instrumentation compliant with NUREG 1.23 (NRC, 1985) is described in the CEC SAR (LES, 1993a). All instrumentation will be located on a tower south of the plant, at least a distance three times the height of the tallest CEC stack, and installed at a height of nearly 40 meters (120 feet). The location of the instrumentation complies with the guide, but the height of installation significantly exceeds the standard installation height of meteorological instrumentation (temperature at 2 meters, wind speed and direction at 10 meters). As a result, CEC site data on temperature and wind speed and direction will be slightly different from data collected under standard weather station conditions. The net effect will be data showing faster but smaller temperature fluctuations at sunrise or sundown, and higher wind speeds with fewer changes of direction than would data collected closer to ground level. Differences between data collected at the CEC site and estimates by models using NWS data should not be significant and should be smaller than the range of error inherent in most model estimates. The data will be logged and processed on a computer in order to generate monthly and annual joint frequency distributions of wind speed and direction as a function of atmospheric stability. Instrumentation will be serviced according to the manufacturers' recommendations and as needed to ensure at least 90-percent data recovery.

2.3.2.2 Normal Release Diffusion Estimates

NRC Regulatory Guide 1.111 (NRC, 1977a) provides guidance for applying Gaussian plume modeling in order to estimate concentrations of material released during normal operation. The models include plume rise, building wake effects, and frequency of occurrence of stability class, wind speed, and wind direction. Joint frequency distributions of concentration per unit source (χ/Q) are calculated using the XOQDOQ computer code (Sagendorf et al., 1982). χ/Q predictions by the NRC staff using this model are summarized in Table 2.8. In addition, the NRC staff used the XOQDOQ model to review the applicant's χ/Q estimates and concludes that the results are consistent and appropriate for use in evaluating normal operational impacts.

Table 2.8 Annual average dispersion analysis (χ/Q) for the CEC (s/m^3)

Direction	Distance (m)									
	805	2414	4023	5632	7241	12068	24135	40225	56315	72405
S	3.428x10^{07}	1.392x10^{07}	8.177x10^{08}	5.723x10^{08}	4.365x10^{08}	2.509x10^{08}	1.157x10^{08}	6.507x10^{09}	4.437x10^{09}	3.330x10^{09}
SSW	1.685x10^{07}	7.659x10^{08}	4.641x10^{08}	3.298x10^{08}	2.539x10^{08}	1.480x10^{08}	6.912x10^{09}	3.903x10^{09}	2.667x10^{09}	2.004x10^{09}
SW	1.579x10^{07}	6.488x10^{08}	3.941x10^{08}	2.819x10^{08}	2.183x10^{08}	1.287x10^{08}	6.085x10^{09}	3.467x10^{09}	2.381x10^{09}	1.795x10^{09}
WSW	1.505x10^{07}	6.569x10^{08}	3.941x10^{08}	2.780x10^{08}	2.128x10^{08}	1.227x10^{08}	5.652x10^{09}	3.167x10^{09}	2.154x10^{09}	1.614x10^{09}
W	2.279x10^{07}	1.074x10^{07}	6.977x10^{08}	5.142x10^{08}	4.044x10^{08}	2.432x10^{08}	1.164x10^{08}	6.656x10^{09}	4.573x10^{09}	3.448x10^{09}
WNW	2.078x10^{07}	1.131x10^{07}	8.429x10^{08}	6.827x10^{08}	5.733x10^{08}	3.851x10^{08}	2.062x10^{08}	1.250x10^{08}	8.868x10^{09}	6.828x10^{09}
NW	3.184x10^{07}	1.879x10^{07}	1.421x10^{07}	1.148x10^{07}	9.594x10^{08}	6.352x10^{08}	3.340x10^{08}	2.003x10^{08}	1.413x10^{08}	1.084x10^{08}
NNW	3.077x10^{07}	1.620x10^{07}	1.164x10^{07}	9.329x10^{08}	7.811x10^{08}	5.243x10^{08}	2.819x10^{08}	1.716x10^{08}	1.221x10^{08}	9.417x10^{09}
N	5.466x10^{07}	2.553x10^{07}	1.652x10^{07}	1.232x10^{07}	9.828x10^{08}	6.102x10^{08}	3.052x10^{08}	1.792x10^{08}	1.251x10^{08}	9.534x10^{09}
NNE	1.904x10^{07}	8.614x10^{08}	5.768x10^{08}	4.422x10^{08}	3.600x10^{08}	2.320x10^{08}	1.205x10^{08}	7.227x10^{09}	5.103x10^{09}	3.918x10^{09}
NE	1.878x10^{07}	7.587x10^{08}	5.000x10^{08}	3.817x10^{08}	3.101x10^{08}	1.996x10^{08}	1.035x10^{08}	6.206x10^{09}	4.380x10^{09}	3.363x10^{09}
ENE	1.755x10^{07}	9.174x10^{08}	7.759x10^{08}	6.787x10^{08}	5.979x10^{08}	4.322x10^{08}	2.466x10^{08}	1.543x10^{08}	1.112x10^{08}	8.654x10^{09}
E	1.593x10^{07}	8.837x10^{08}	7.999x10^{08}	7.303x10^{08}	6.607x10^{08}	4.969x10^{08}	2.931x10^{08}	1.860x10^{08}	1.351x10^{08}	1.056x10^{08}
ESE	1.307x10^{07}	6.298x10^{08}	4.489x10^{08}	3.611x10^{08}	3.040x10^{08}	2.070x10^{08}	1.131x10^{08}	6.948x10^{09}	4.967x10^{09}	3.845x10^{09}
SE	1.836x10^{07}	7.362x10^{08}	4.254x10^{08}	2.934x10^{08}	2.213x10^{08}	1.248x10^{08}	5.639x10^{09}	3.135x10^{09}	2.125x10^{09}	1.588x10^{09}
SSE	1.537x10^{07}	5.542x10^{08}	3.049x10^{08}	2.067x10^{08}	1.551x10^{08}	8.783x10^{09}	4.041x10^{09}	2.286x10^{09}	1.566x10^{09}	1.180x10^{09}

2.3.2.3 Accident Diffusion Estimates

Regulatory Guide 1.145 (NRC, 1982b) provides guidance for applying Gaussian plume modeling of continuous releases and selecting meteorological conditions representative of accident conditions, including plume rise, building wake effects, and frequency of occurrence of stability class, wind speed, and wind direction. χ/Q estimates were calculated for each of 16 direction categories and for all directions taken together, for both elevated and ground-level releases.

For elevated releases, a summary of accident condition dispersion modeling estimates of χ/Q for a set of distances is presented in Table 2.9. The results indicate that the maximally exposed individual is located in the northern sector at a distance of 400 meters and that the 95-percent overall and the largest 99.5-percent sector χ/Q estimates are approximately equal at 1.7×10^{-5} s/m^3.

For continuous ground-level releases, accident condition dispersion modeling estimates of χ/Q are summarized in Table 2.10. The distances selected for these estimates are locations of potentially maximally exposed individuals. The first individual is located at the restricted area fence; the second, at the point of maximum exposure located for elevated releases (400 meters). In this case, the east is the direction of maximum χ/Q, and the 95-percent overall and the largest 99.5-percent sector χ/Q estimates are approximately equal. χ/Q values for the fence-line and 400-meter receptors are 6.9×10^{-3} and 1.5×10^{-3} s/m^3, respectively.

2.4 Hydrology

Surface water flowing from the CEC site is a potential pathway for transporting radioactive materials to local residents and the environment. In addition, radionuclides deposited onto soils or sediments could be leached into groundwater and transported to potential human receptors. Radionuclides released from the CEC stacks could be a source for surface soil contamination. Using a conservative estimate of annual atmospheric releases for uranium (120 µCi, SER Chapter 7), the maximum annual average χ/Q (see SER Table 2.8), and a deposition velocity of 0.001 m/s (Napier, 1990), the NRC staff estimated that uranium deposition for a 30-year period would be 2.0×10^{-4} pCi/cm^2. If this quantity of uranium were dispersed through the upper centimeter of soil, the average uranium concentration would be 1.0×10^{-4} pCi/g. This is a small fraction of normally occurring uranium concentrations in soil. Similarly, if all of the uranium contained in a 30-year volume of CEC liquid effluent were to accumulate in a 1-centimeter layer of Bluegill Pond sediment, uranium concentration would be 0.4 pCi/g. Given the conservative assumptions and the low solubility of uranium in water, this is an insignificant concentration. On the basis of these considerations, the NRC staff concludes that potential CEC releases to groundwater have insignificant public health and safety impacts. A description of the groundwater resources is presented in the CEC DEIS (NRC, 1993a). This section describes surface water discharge from the CEC site, characterizes potential exposure pathways, and estimates the potential for flooding.

Table 2.9 Frequency distributions of concentration per unit source, for elevated releases

Direction	χ/Q (s/m^3) Distance to Receptor		
	200 m	400 m	600 m
95% Overall			
	$8.4X10^{-6}$	$1.7x10^{-5}$	$1.3x10^{-5}$
99.5% Sector			
S	$8.1x10^{-6}$	$1.7x10^{-5}$	$1.2x10^{-5}$
SSW	$1.8x10^{-6}$	$1.1x10^{-5}$	$9.5x10^{-6}$
SW	$1.8x10^{-6}$	$1.1x10^{-5}$	$1.0x10^{-5}$
WSW	$1.7x10^{-6}$	$1.1x10^{-5}$	$9.7x10^{-6}$
W	$6.0x10^{-6}$	$1.4x10^{-5}$	$1.1x10^{-5}$
WNW	$1.8x10^{-6}$	$1.3x10^{-5}$	$1.1x10^{-5}$
NW	$7.2x10^{-6}$	$1.5x10^{-5}$	$1.1x10^{-5}$
NNW	$5.5x10^{-6}$	$1.4x10^{-5}$	$1.1x10^{-5}$
N	$9.2x10^{-6}$	$1.7x10^{-5}$	$1.4x10^{-5}$
NNE	$5.3x10^{-6}$	$1.3x10^{-5}$	$1.1x10^{-5}$
NE	$7.1x10^{-6}$	$1.4x10^{-5}$	$1.1x10^{-5}$
ENE	$7.0x10^{-6}$	$1.5x10^{-5}$	$1.1x10^{-5}$
E	$6.8x10^{-6}$	$1.4x10^{-5}$	$1.1x10^{-5}$
ESE	$1.7x10^{-6}$	$1.1x10^{-5}$	$9.6x10^{-6}$
SE	$1.8x10^{-6}$	$1.1x10^{-5}$	$9.9x10^{-6}$
SSE	$1.8x10^{-6}$	$1.1x10^{-5}$	$9.9x10^{-6}$

Table 2.10 Frequency distributions of concentration per unit source, for ground-level releases

Direction	χ/Q (s/m^3) Distance to Receptor	
	165 m	400 m
95% Overall		
	$6.9X10^{-3}$	$1.5x10^{-3}$
99.5% Sector		
S	$1.9x10^{-3}$	$4.8x10^{-4}$
SSW	$1.3x10^{-3}$	$2.8x10^{-4}$
SW	$1.2x10^{-3}$	$2.7X10^{-4}$
WSW	$1.2x10^{-3}$	$2.7x10^{-4}$
W	$2.3x10^{-3}$	$5.7x10^{-4}$
WNW	$6.5x10^{-3}$	$1.5x10^{-3}$
NW	$7.4x10^{-3}$	$1.6x10^{-3}$
NNW	$7.4x10^{-3}$	$1.6x10^{-3}$
N	$7.0x10^{-3}$	$1.6x10^{-3}$
NNE	$4.0x10^{-3}$	$8.8x10^{-4}$
NE	$3.4x10^{-3}$	$7.5x10^{-4}$
ENE	$7.4x10^{-3}$	$1.6x10^{-3}$
E	$7.8x10^{-3}$	$1.7x10^{-3}$
ESE	$4.1x10^{-3}$	$9.1x10^{-4}$
SE	$1.2x10^{-3}$	$2.8x10^{-4}$
SSE	$5.1x10^{-3}$	$1.2x10^{-4}$

2.4.1 Regional Surface Water Hydrology

The CEC site is located in the western portion of the Ouachita River Basin, as shown in Figure 2.7. The Ouachita River drains to the east and flows into the Mississippi River, which flows south into the Gulf of Mexico. Major tributaries include Bayou D'Arbonne, which is located to the east and south of the site and flows southeast into Lake Claiborne. Its branches flow from northwest to southeast into the Ouachita River. The Middle Fork of Bayou D'Arbonne is located northeast of the site and flows southeast into Bayou D'Arbonne Lake.

The largest surface water body in the vicinity of the site is Lake Claiborne, a man-made lake created for flood control in 1966 by damming Bayou D'Arbonne. This lake is used extensively for recreational purposes, including boating, fishing, swimming, and water skiing, but not as a source of public drinking water.

2.4.2 Local Surface Water Hydrology

The CEC site is located on a drainage divide (LES, 1993b). Surface water flows from the site to the northeast, northwest, and southwest. The surface water resources of the CEC site and vicinity include small streams and two man-made lakes, Bluegill Pond and Lake Avalyn, as shown in Figure 2.8. Surface water flowing from the site to the west and northwest discharges to Cypress Creek, which joins Beaver Creek just before it flows south into Lake Claiborne. Surface water flowing to the east from Lake Avalyn forms the headwaters of McCasland Creek, which eventually discharges into the Middle Fork of Bayou D'Arbonne, a Louisiana Natural and Scenic State Stream (Emmer et al., 1983). An unnamed stream has its headwaters in the extreme southeast corner of the site and flows southwest to Cypress Creek and then into Lake Claiborne.

The onsite streams are generally intermittent but have flow rates recorded up to a few cubic feet per second (LES, 1993b). Onsite stream flows are generally an order of magnitude greater in January than in July and August. Recently obtained streamflow measurements of streams on the CEC site are listed in Table 2.11.

The two main streams which drain the LES site are Cypress Creek and McCasland Creek. Cypress Creek flows south into Bayou D'Arbonne, and McCasland Creek flows east into the Middle Fork of Bayou D'Arbonne. In August 1990, the downstream reaches of these two creeks were dry. Human use of Cypress and McCasland Creeks was not documented; however, it is possible that children in the area may play in and on the banks of these streams. In addition, livestock raised by residents living along the downstream reaches of both creeks may use the streams as sources of water.

The two man-made lakes on the site, Bluegill Pond and Lake Avalyn, have surface elevations of 85 and 90 meters (275 feet and 297 feet), respectively. Bluegill Pond has a drainage basis of 1.05 hectares (2.6 acres) Lake Avalyn of 68.4 hectares (169 acres). On the basis of depth

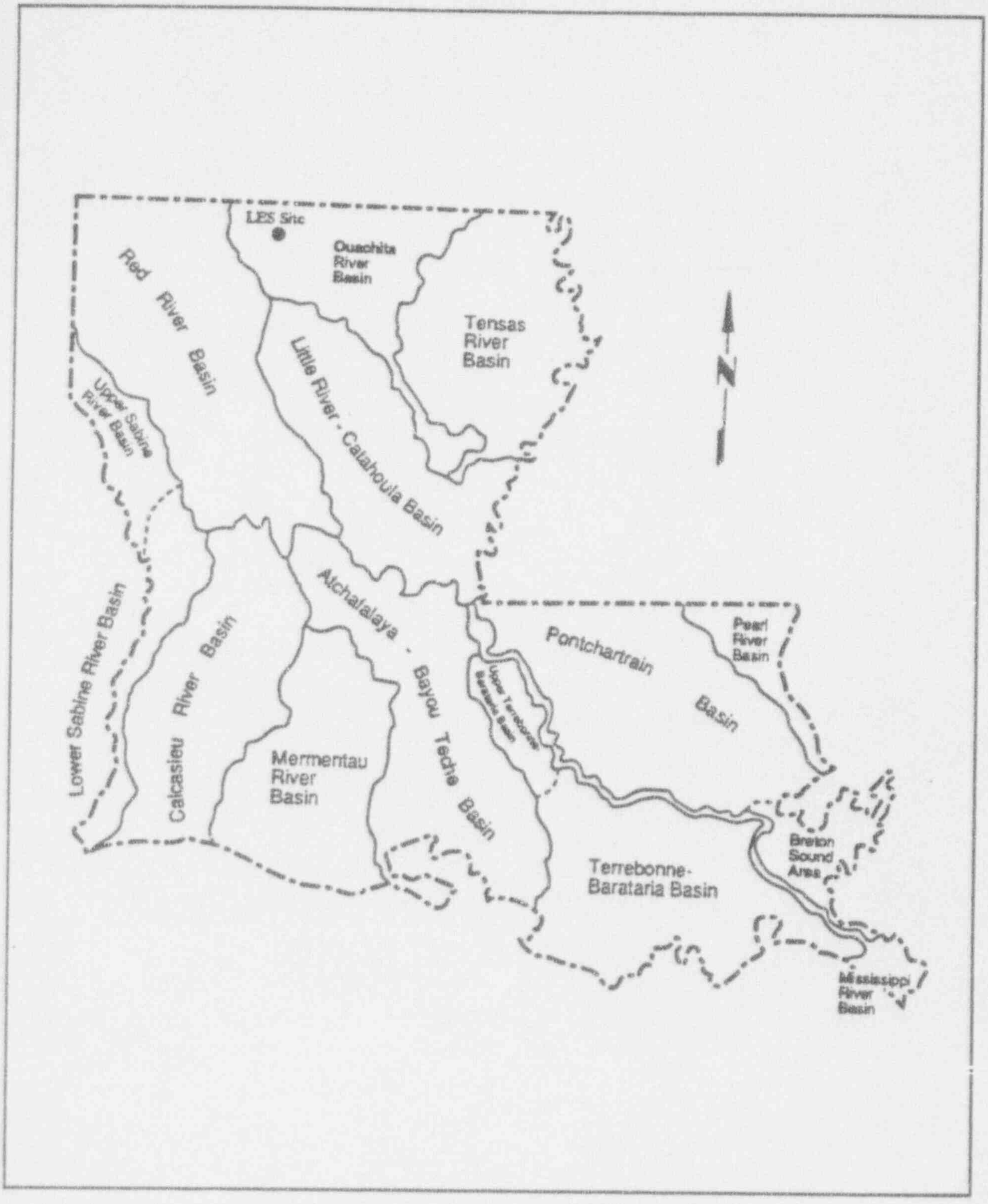

Figure 2.7 Map of Louisiana river basins (Emmer et al., 1983)

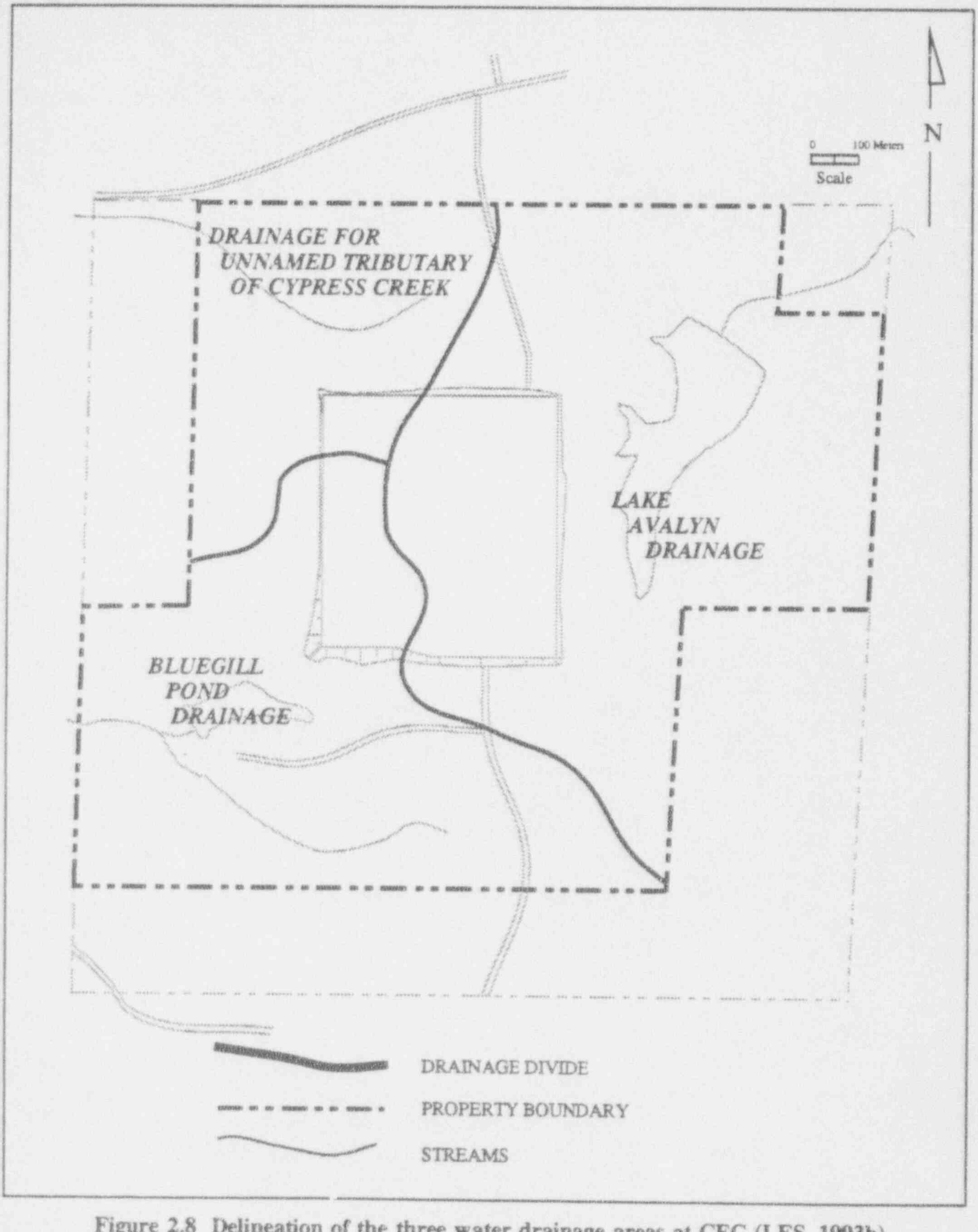

Figure 2.8 Delineation of the three water drainage areas at CEC (LES, 1993b)

Table 2.11 Estimates of surface water flow in streams on or in the immediate vicinity of the CEC site

Location	Discharge (m^3/sec) January 1990	May 1990	July 1990
Lake Avalyn Drainage Basin			
Southern flow to Lake Avalyn	0.017	0.005	NE
Discharge from Lake Avalyn	0.068	0.047	0.002
Bluegill Pond Drainage Basin			
Total flow into Bluegill Pond	0.013	0.011	NE
Discharge from Bluegill Pond	0.019	0.014	NE
Flow in tributary from SW corner of CEC site to CEC site	0.040	0.016	NF
Flow at the SW site boundary after confluence of tributary from the SW and Bluegill Pond discharge	0.054	NE	0.006
Northwest Drainage Basin			
Flow in tributary on NW corner of LES property	0.009	NE	NF

Source: LES, 1992a
NE = not estimated.
NF = no flow identified; standing water only.

surveys, Bluegill Pond has a volume of 19,820 cubic meters (700,000 ft^3); Lake Avalyn, a volume of 113,250 cubic meters (4,000,000 ft^3) (LES, 1993b). Discharge from Bluegill Pond was less than 0.03 cubic meters per second (1 ft^3/sec) in both January and May 1990. Discharge from Lake Avalyn was 0.07 cubic meters per second (2.4 ft^3/sec) in January 1990, and 0.05 cubic meters per second (1.65 ft^3/sec) in May 1990 (Table 2.10). Because of the wide variability in flows and for a conservative impact analysis, the NRC staff used the applicant's estimate of annual average developed area run-off of 0.012 cubic meters per second.

The Army Corps of Engineers (COE) determined that a small area of wetlands exists on the proposed site (Westinghouse, 1989). This area is located in the northeast corner of the site, downstream from the CEC facility and the Lake Avalyn dam. The soils map identifies this area as Iuka-Darley soils, which are subject to flooding by local small streams.

2.4.3 Flooding Potential

The applicant evaluated (LES, 1993a) the flooding potential at the CEC site based on the following sources of information: a COE site assessment and an examination of flood insurance maps, maps developed by the U.S. Soil Conservation Service (SCS) (DOA, 1956), and U.S. Geological Survey (USGS) topographic maps. Estimates of the probable maximum precipitation and the design storm are based on data provided by the National Oceanic and Atmospheric Administration (NOAA) and the SCS. The results of these evaluations are discussed in this section.

The COE assessed the flooding potential of the CEC site (COE, 1989) and determined that the proposed site is located in an area of minimal flooding. The USGS topographic map for the area indicates that the proposed site is on a topographic high, is not located in or near the floodplains of any major streams or rivers, and is not adjacent to any major bodies of water. Finally, an applicant search for flood insurance maps of the area found that none has been issued for the site area--a suggestive but unreliable indicator that flooding in the area is not a concern.

Because the site is located on the top of a hill, the applicant proposed that the design basis flood for the site results from locally intense precipitation on the site (LES, 1993a). The Design Basis Flood Level (DBFL) is the Standard Project Flood (SPF), as defined by the COE (LES, 1993a). Rainfall from this storm amounts to 15 centimeters (6 inches) over a 6-hour period. By assuming that the edge of the developed area acts as a weir, the applicant estimated that up to 6.2 centimeters (2.5 inches) of water could pond on the facility grounds during the storm.

The NRC staff reviewed the basis and method for estimation of the DBF and concludes that the results are reasonable and consistent with the requirements of the ANPR guidance for selection of design basis natural phenomena events.

2.5 Geology and Seismology

The applicant reported geologic, seismologic, and other geotechnical data, including foundation and earthquake information for consideration in the design and construction of the CEC, which compose a portion of the CEC design basis (Law Engineering Services, 1990a, 1990b). This section describes the geologic and seismologic data and analysis on which the CEC design is based in light of data in other relevant technical documents. It also assesses the CEC design and construction in terms of the relevant geologic, seismologic, and other geotechnical criteria.

Appropriate design and construction criteria for buildings subject to earthquake were developed in the Building Seismic Safety Council (BSSC, 1988) National Earthquake Hazards Reduction Program (NEHRP). The purposes of the provisions are to (1) minimize the hazard to life for all buildings, (2) increase the expected performance of higher-occupancy structures and (3) improve the capability of essential facilities to function during and after an earthquake. The provisions are expected to "provide the minimum criteria considered to be prudent and economically justified for protection of life safety in buildings subject to earthquakes at any location in the United States" (BSSC, 1988, p. 1).

In its reports, the applicant concludes:

1. Faulting in the area of the proposed site is related to regional subsidence of the salt basin and to salt intrusion. These faults are considered to be inactive because the middle tertiary faults do not exist at the site.

2. Although the site is located on the northwestern flank of the Homer salt pillar (an incipient salt dome), the salt pillar ceased its structural development in the middle to late Cretaceous.

3. The site and surrounding region have a history of very few earthquakes.

4. Three design-basis earthquakes were analyzed for groundshaking at the site:
   (a) The design earthquake for a far-field event was chosen to be an $m_b = 6.7$ earthquake located at a distance of 365 kilometers (225 miles) from the point in the New Madrid Fault Zone closest to the site. This earthquake magnitude was chosen because it represents the statistical 500-year return period for this earthquake source zone. The largest earthquake known to have occurred in the fault zone is the 1812 earthquake, with $m_b = 7.3$.
   (b) The design earthquake for a mid-field event was chosen to be an $m_b = 5.7$ earthquake occurring 105 kilometers (65 miles) from the site and represents the statistical 500-year return period.
   (c) The design earthquake for a near-field event was chosen to be an $m_b = 4.3$ earthquake occurring 14 kilometers (9 miles) from the site and represents the statistical 500-year return period.

5. The possibility of liquefaction can occur in Stratum IV and Stratum V (silts, fine sand with some clay beds). Some of these soils are above the water table and, thus, are not susceptible to liquefaction.

6. Compaction of soils resulting from earthquake groundshaking can occur in Stratum IV and Stratum V.

The first subsection of this geology and seismology review summarizes the basic data presented by the applicant. The second subsection summarizes the applicant's analysis along with the NRC staff review of the applicant's analysis. The final subsection summarizes the NRC staff review. A major element of the NRC staff review is an evaluation of the characteristics of the CEC design basis earthquake (DBE).

2.5.1 Basic Geologic Information

The applicant conducted literature searches and reviewed surface and subsurface studies of regional and site-specific geology, site-specific soils engineering, and restricted site-specific geophysics (cross-hole/down-hole seismic) (Law Engineering Services, 1990a; 1990b) in order to determine the geologic and geotechnical suitability of the site for the CEC facility. Topics of regional studies included physiography, stratigraphy, tectonic setting, fault activity, salt domes, and mineral resources. The applicant conducted no field investigations (e.g., geologic mapping of reported active or suspected active faults) and provided no new information or interpretations of the regional geologic data. Within these limits, the applicant described the regional and site-specific tectonic setting, fault activity, and salt domes and used this information to define areas or sources of potential earthquakes which might affect the site for the far-field, mid-field, and near-field earthquakes. The NRC staff did not inspect the site, excavations, or soils retrieved from borings.

2.5.1.1 Regional Geology

The proposed site in northern Louisiana is within the Interior Salt Basin Region of the Gulf Basin or Gulf Coast Geosyncline. The Geosyncline is composed of Jurassic to Quaternary sedimentary rocks, primarily shale, limestone, sandstone, anhydrite, and silt.

Within the Geosyncline, the rocks have been subjected to local broad uplift, have been locally folded, and are offset by normal, dip-slip faults. Most of the normal faults are considered to have been caused by differential subsidence of the Geosyncline. In addition, some of the salt beds have deformed and developed salt domes. Faults are associated with these salt domes.

Major tectonic regions within a 320-kilometer (200-mile) radius of the site include: (1) Interior Salt Basin Region, (2) Gulf Coast Region, (3) Central Texas Region, (4) Ouachita Region, (5) Wichita-Arbuckle Region, (6) Reelfoot Rift, (7) New Madrid Fault Zone, (8) Central Stable Region, and (9) Mississippi Embayment. Each of these regions has different tectonic characteristics, and some have greater potential for producing major earthquakes than other regions. Some of these regions have faults attributed to tectonic origins (e.g., Ouachita Region, Reelfoot Rift, and New Madrid Fault Zone), whereas other regions have faults attributed to non-tectonic origins (e.g., the growth faults of the Mississippi Embayment or faults associated with the development of salt domes). In some areas, faults are attributed to extraction of fluids (e.g., oil, water) from the subsurface.

2.5.1.1.1 Physiographic Setting

The physiographic setting of northern Louisiana is generally flat. Average elevation of the upland areas is approximately 90 meters (300 feet) above MSL. Major drainage features are the southward-flowing Mississippi River to the east and the Red River to the west. These rivers occupy broad floodplains several miles wide with steep sides. Topographic relief averages approximately 30 meters (100 feet). These physiographic features were influenced by the lowering of sea level during the Quaternary. Within the last 18,000 years, a slight rise in sea level has decreased the rate of erosion.

2.5.1.1.2 Regional Structural Geologic Setting

The present regional structural geologic setting began to develop during the late Paleozoic when the incipient Gulf Basin or Gulf Coast Geosyncline formed by differential subsidence of the continental crust. Block faulting accompanied the subsidence. Erosion of the higher areas occurred during the Mesozoic (Triassic). Differential subsidence continued, and the lower areas (basins) filled with detrital materials from the erosion and with evaporite deposits such as salt and anhydrite. At the proposed site in northern Louisiana, one of these basins is termed the North Louisiana Salt Basin, or the North Louisiana Syncline. The block faulting during the Triassic produced two ancient, high areas adjacent to the Louisiana Salt Basin: the Sabine Uplift in northwestern Louisiana and the Monroe Uplift in northeastern Louisiana. The Louisiana Salt Basin is bounded on its north by the South Arkansas Fault Zone, which is a series of faults. Northeast of the South Arkansas Fault Zone are the New Madrid Fault Zone and Reelfoot Rift. Law Engineering Services considers the New Madrid Fault Zone as the likely source of the largest earthquake which might affect the site; the NRC staff agrees. West of the site are faults along the edge of the Ouachita Region. Other major faults or fault zones include the Balcones, Luling, Mexia-Talco, and Charlotte-Jordonton.

Growth faults are considered to be non-tectonic, gravity-related features formed contemporaneously with sediment deposition and the downwarping of the Gulf of Mexico. These faults are characterized by steep, near-surface dips, which become less steep with depth and eventually pass into bedding planes at great depth. Movement on most known growth faults ceased in the Tertiary. As a result of subsequent deposition, these faults are under high lithostatic stress and, thus, have little potential for surface displacement. Some have continued to move, principally as a result of human activity (e.g., the withdrawal of groundwater and fluids associated with hydrocarbon production). The applicant has not identified such active growth faults in the immediate area of the proposed site.

2.5.1.1.3 Stratigraphy

Since the Jurassic, sedimentary rocks filled the basins, essentially continuously, as the region subsided. An unconformity occurs between the Miocene and Pleistocene rocks. The Pleistocene and recent sedimentary deposits are primarily sands, silts, and clays derived from existing nearby sedimentary rocks.

2.5.1.1.4 Economic Resources

Economic resources of the North Louisiana Basin include oil, gas, salt, sulfur, lignite, iron ore, and native materials used for constructions, such as soil, sand, and gravel. Oil and gas comes primarily from Jurassic through Tertiary rocks. Salt and sulfur are extracted from the salt domes which pierced the ground surface. Lignite is removed from the Tertiary Wilcox group, and iron ore comes from glauconitic deposits.

2.5.1.2 Site Geology

The geology at the surface is relatively simple, but the geology in the near subsurface indicates some complexities and indications of structure. On the basis of information in the Law Engineering Services report, the NRC staff concludes that known geologic structures at or in the immediate vicinity of the site do not present a known safety hazard to the proposed facility. Although there is potential for liquefaction and compaction, it is low, and prudent engineering and construction should minimize their potential impacts.

2.5.1.2.1 Physiographic Setting

The site is approximately 104 meters (340 feet) above MSL in the central areas and 85 meters (280 feet) above MSL to the south. The site contains gently rolling hills within the Ouachita River drainage basin. Surface drainage flows both to the south and west, and to the east. The topography between the creeks is relatively flat and forms a drainage plain. Surface drainage to the southwest flows into Cypress Creek and eventually into Lake Claiborne. Surface drainage to the east is into Lake Avalyn, which is drained by McCasland Creek. Vegetation at the site is extensive and consists of pine and oak trees. Where trees have been cut, the remaining trunks stand approximately 15 centimeters (6 inches) above the ground.

2.5.1.2.2 Site Structural Geologic Setting

The site is located between the north flank of the North Louisiana Basin and the southwest flank of the Monroe Uplift. Soil borings indicate that the rocks at the site have a low dip to the southwest. Faults in the general area are related to regional subsidence and to intrusion of salt domes. The applicant believes that faulting has not been active since the middle Tertiary because sedimentation has not occurred since then; basin subsidence and salt dome growth have not occurred because sedimentation has ceased.

The applicant reports that salt domes do not exist at the site. The site, however, overlies the northeast flank of the Homer salt pillar, an incipient dome, which reportedly developed during the late Jurassic to early Cretaceous. It apparently did not develop further during the middle to late Cretaceous.

A northeast-striking fault was reported under the site at a depth of 1,800 meters (5,900 feet) below the surface with a vertical offset of 30 meters (100 feet) (Law Engineering, 1981). The fault is below apparently unfaulted Cretaceous rocks.

2.5.1.2.3 Stratigraphy

The site is underlain by rocks that range in age from lower Jurassic to Tertiary. The thickness of these rocks is 6,100 meters (20,000 feet), and they overlie rocks of the Triassic basement. The Tertiary rocks are overlain by a relatively thin veneer of recent alluvial deposits.

2.5.1.2.4 Economic Resources

Economic resources developed near the site are oil, gas, and native materials, such as soil and sand. Oil and gas are produced 16 kilometers (10 miles) to the southwest.

2.5.1.3 Site Exploration

The applicant explored the site to determine the thickness and geotechnical characteristics of the near-surface soils, groundwater table and perched water, and in-situ elastic properties for seismic analysis. These data were obtained from soil borings, test pits, electric cone penetrometer tests (CPT), and down-hole and cross-hole geophysical surveys. The applicant also conducted laboratory tests of the soils. Soils engineering and construction recommendations were based on data analysis and interpretation.

2.5.1.3.1 Location Surveys

The applicant conducted field surveys for locating soil test borings and electric CPTs by using an established reference point at the northwest corner of the property.

2.5.1.3.2 Soil Test Borings

Approximately 54 test borings were completed by using two all-terrain vehicles and rotary wash drill rigs. Samples were collected at 0.6-meter (2-foot) intervals in the upper 3 meters (10 feet) and at 1.5 meter (5-foot) intervals below 3 meters (10 feet). The depth of drilling ranged from 8 to 30 meters (25 to 100 feet). Standard penetration tests were conducted. Soil samples were collected by using thin-wall tubes and pitcher samplers, as appropriate. All tests followed American Society of Testing Materials (ASTM) specifications. Four of these borings were converted for use in measuring groundwater levels by installing temporary piezometers. Water levels were measured daily. All borings were logged and the soils described according to the Unified Soil Classification System.

2.5.1.3.3 Electric Cone Penetrometer Tests

CPTs were conducted at fifteen locations within the process area to determine the in-situ point resistance, side friction, and friction ratio of the soils. The tests were conducted according to ASTM requirements. The field data were related by computer to equivalent blow counts, friction angle, and undrained shear strength.

2.5.1.3.4 Test Pits

Thirteen test pits were excavated by backhoe to depths of 3 meters (10 feet) to evaluate the soils collected as disturbed bag samples for suitability as yard fill and backfill. Samples were collected according to ASTM requirements.

2.5.1.3.5 Geophysical Surveys

Two down-hole and two cross-hole seismic tests were conducted to obtain in-situ seismic velocities and Poisson's ratio with depth in the subsurface soils. The two down-hole surveys reached depths of 30.3 and 30.7 meters (99.4 and 100.7 feet), and the two cross-hole surveys reached depths of 13.9 and 30.4 meters (45.6 and 99.7 feet).

Grout, the water table, and perched water were present in the zone of measured compressional (P) and shear (S) waves in the boreholes. Grout can make the arrival of P-waves difficult to interpret, and the difference in P-wave velocity may possibly result from saturation of the soil from rain (Law Engineering Services, 1990a, p. 37).

The NRC staff notes that the Poisson's ratio values are very high (greater than 0.45) in boreholes B-17 and B-27 (both down-hole and cross-hole) and B-15 (down-hole only). These values occur both above and below the water table. The maximum theoretical value for Poisson's ratio is 0.5; characteristic values of the ratio for unsaturated soils of the types described at the site should be approximately 0.25 to 0.33. Values of the ratio in saturated soils are questionable because of the influence of the water on the velocities of both the P- and S-waves. The NRC staff considers that the samples with calculated Poisson's ratio values greater than 0.45 are suspect and probably not representative of the soil or rock. These values, including the estimated values corrected for changes in stress as a result of grading at the site, probably should not be used in the analyses related to design (Law Engineering Services, 1990b, Table 4-2). An evaluation of this NRC staff finding is presented in SER Section 2.5.3.

2.5.1.3.6 Laboratory Testing

The laboratory tests consisted of determining Atterburg limits, moisture content, unit weight, grain size, triaxial compression, consolidation, permeability, shrink-swell, acidity (pH), resistivity, expansion, standard Proctor, California Bearing Ratio, and resonant column tests. The resonant column tests were conducted by McClelland Engineers.

2.5.2 Vibratory Ground Motion and Ground Response Spectra

The site is subject to vibratory ground motion generated by earthquakes from different seismic sources at different distances. The characteristics of the ground motion at the site depend primarily on the magnitude of the earthquake, its distance from the site, and soils and geologic conditions at the site. These three factors are integrated in establishing the seismic design and construction of buildings.

Several accepted approaches in determining these factors and their application to a site have been developed. The history of earthquakes and their magnitudes in the region is determined. These earthquakes may be related to known geologic structures (e.g., individual faults) or to tectonic regions (e.g., New Madrid Fault Zone). These data are analyzed (e.g., for each tectonic region within a 320-kilometer [200-mile] radius of the site), and a decision is made about the probability of an earthquake of a certain magnitude within a designated period of time.

For example, the data might indicate that a magnitude 6.5 earthquake has a statistical mean recurrence interval of 500 years. This recurrence interval approximates a 90-percent probability that an earthquake of greater magnitude would not occur in a 50-year interval or at an annual risk of 0.002 events per year. Such a magnitude and probability, if acceptable, would be used as the design earthquake. However, the magnitude 6.5 may not be the largest possible magnitude in the tectonic region, but a larger earthquake would have a probable longer recurrence interval; the cost of construction related to design against the larger earthquake might not be warranted by the reduced annual risk of that earthquake.

The expected ground shaking at the site is based on the design earthquake. Earthquake ground shaking is based on field measurements (seismograms) from which specific ground motion parameters are obtained. These parameters are frequency content of the seismic wave; particle acceleration, velocity, and displacement; and duration of ground shaking. If possible, a family of earthquakes representative of the design earthquake is chosen, preferably from the same tectonic region as the design earthquake, and the ground motion parameters of acceleration, velocity, displacement, and frequency (or its inverse, period) are plotted on tri-partite logarithmic paper to develop a ground response spectrum, which depicts the intensity and frequency of the ground motion. The earthquakes used in the development of the spectrum may be some distance from the site, and the parameters have to be attenuated to the site. Attenuation is commonly done by using empirically derived attenuation curves for the respective parameters. The geologic conditions at the site are considered, and the attenuated parameters are adjusted for amplification which might result because of local soil or rock conditions.

This area of the southeastern United States does not have a history of frequent earthquakes of large magnitudes. Thus, the reliability of recurrent events is not high and groundshaking parameters may not be readily available. In such cases, ground motion parameters are calculated by using Random Vibration Theory (RVT).

The different tectonic regions occur at different distances from the site. Each region has the potential for generating earthquakes of different magnitudes and rates of occurrence. In addition, the high-frequency earthquake waves attenuate more rapidly than low-frequency waves, which dominate earthquake effects. The ground response spectra should show these differences.

One deficiency of the ground response spectra is that it does not include a measure of the duration of ground shaking. The major effect of duration of the shaking is its adverse effect on the structural strength of a facility. Generally, for the purpose of design, structural engineers assume a duration of 20 to 30 seconds, but they can incorporate longer durations.

The NRC staff reviewed the applicant's methodology for developing the basis of design for the proposed facility. The applicant used RVT to develop a ground response spectra for the site. Its approach identified the tectonic regions within a 320-kilometer (200-mile) radius of the site; analyzed the earthquakes in these regions and, where possible, related the earthquakes to known faults; identified design earthquakes in the far-field, mid-field, and near-field; and developed site-specific response spectra for each of the design earthquakes. The NRC staff considers the methodology to be reasonable and consistent with state-of-the-art practice. The analyses incorporated available data on the seismic history in the tectonic regions; established design earthquakes for the far-field, mid-field, and near-field; attenuated the ground motion parameters to the site; incorporated the local (site) geologic conditions; developed site-specific ground response spectra; and developed a combined spectra. On the basis of review of the data presented in the applicant's reports, the NRC staff concludes that the applicant's method is reasonable and conservative.

2.5.2.1 Requirements for Seismic Design Basis

The ANPR specified that the design basis earthquake have a return period on the order of 500 years and referenced the NEHRP risk maps (BSSC, 1988) for identification of the vibratory ground motion. NEHRP facilities designed to this criteria are identified as Seismic Hazard Exposure Group III (BSSC, 1988, p. 5); facilities in this group have the highest level of protection and design performance because they would function during and after, and recover afterwards from, an earthquake.

Ground motion produced at the site by the NEHRP Group III design earthquake has a 90-percent probability of not being exceeded in 50 years. The ground response spectra is to be based on this design earthquake. However, the scope of this section does not include the structural design of the facility, the application of the ground response spectra to it, the soil/structure interaction, vertical distribution of seismic forces in the facility, and other properties of the facility's response to earthquakes.

2.5.2.2 Analysis of Law Engineering Services Seismic Design Basis

The applicant's methodology in analyzing ground motion characteristics is standard and state-of-the-art. The approach identified discrete seismic source zones (faults, tectonic regions) and assessed the maximum earthquake magnitude for each zone. The temporal occurrence of earthquakes was determined for the zones, and ground motion parameters were attenuated to the site. Because there are few earthquakes generated in the source zones, the RVT was used to compute directly the theoretical response spectra as well as to provide values of peak particle acceleration, velocity, and displacement.

Seismic data for the NEHRP 475-year earthquake are obtained from the hazard analysis developed in the NEHRP methodology. The data can come from earthquakes of different size and distance from the site. The estimate of the earthquake size is based on the history of earthquakes in the different seismogenic regions. The applicant presented three design earthquakes: (1) a small, near-field event; (2) a moderate, mid-field event; and (3) a large, far-field earthquake occurring in the New Madrid Fault Zone. The applicant developed a ground response spectra that combined the information from all three spectra.

2.5.2.2.1 Seismicity and Seismotectonic Zones

The applicant's analysis concluded that the tectonic and seismotectonic regions within 320 kilometers (200 miles) of the site are: (1) Interior Salt Basin Region, (2) Gulf Coast Region, (3) Central Texas Region, (4) Ouachita Region, (5) Wichita-Arbuckle Region, (6) Reelfoot Rift, (7) New Madrid Fault Zone, (8) Central Stable Region, and (9) Mississippi Embayment. Each region is briefly described below.

(1) Interior Salt Basin Region. This tectonically stable region contains the Gulf Coast Basin boundary fault system, the Angelina-Caldwell flexure, the Sabine and Monroe Uplifts, and several basins with salt domes. Six earthquakes with magnitudes greater than 3.5 have been reported from the region, the largest of which was 4.1, located beyond the 320-kilometer (200-mile) radius of the site. Some relatively small earthquakes have been interpreted as being related to hydrocarbon production. The maximum earthquake assigned to the region is 4.9.

(2) Gulf Coast Region. The offshore region is experiencing slow subsidence and is characterized by active growth faults; the applicant considers these faults to be aseismic. Seismicity of the region is low. Four earthquakes with magnitudes between 4.2 and 4.4 have been reported but have not been related to known geologic structures. A maximum earthquake of 4.9 has been assigned to the region.

(3) Central Texas Region. Seismicity in this tectonically stable region is low. The region is bounded on the east by the Rio Grande Rift, on the south by the Wichita-Arbuckle Uplift, and on the west by the Mexia-Talco Boundary Fault

Zone. Major structures include the Llano Uplift and the Permian Basin. The applicant considers the surface faults in the region to be inactive. Eight earthquakes with magnitudes from 4.0 to 4.6 have been reported. A maximum magnitude of 5.7 has been assigned to the region.

(4) Ouachita Region. The region is relatively active seismically and tectonically. Many surface and subsurface faults are known, but earthquakes have not been directly related to these faults and are widely distributed throughout the region. The largest historic earthquake had a magnitude of 4.8. A maximum earthquake of 5.7 has been assigned to the region.

(5) Wichita-Arbuckle Region. Seismicity in the region is relatively active. Structures include the Wichita Uplift, the Arbuckle Mountains, and the Muenster Arch. The Meers Fault has inferred Quaternary displacement and a magnitude 6.1 to 6.6 earthquake has been inferred as possible. Five earthquakes with magnitudes between 4.0 and 4.8 have been reported. A magnitude 6.8 has been assigned to the region.

(6) & (7) Reelfoot Rift and New Madrid Fault Zone. These two apparently connected regions are considered to have the potential for producing the largest earthquakes in the area. The Reelfoot Rift has had four earthquakes between 4.5 and 4.9. A maximum magnitude 6.8 has been assigned. The New Madrid Fault Zone has had historic earthquakes between 7.0 and 7.4 in 1811-1812. Many other earthquakes are known to have occurred in the region. The maximum earthquake assigned to the region is 7.4.

(8) Central Stable Region. The region is considered stable, with low seismicity. Major structures include the Anadarko Basin and a portion of the Nemaha Ridge. Four earthquakes between 4.4 and 5.5 have been reported. The maximum magnitude assigned is 5.7.

(9) The Mississippi Embayment. The region is moderately seismically active with 13 earthquakes reported between 4.5 and 5.4. The maximum magnitude assigned is 5.7.

2.5.2.2.2 Correlation of Seismic Activity with Geologic Structures

The applicant correlated the seismic activity within each seismotectonic region with individual geologic structures. This relation is summarized in Table 2.12.

Table 2.12 Summary of seismic activity related to geologic structures

Region	Structure	Seismic Activity (R) Region (S) Structure	Number of Earthquakes (R) Region (S) Structure	Maximum Magnitude Assigned (R) Region (S) Structure
Interior Salt Basin	• Gulf Coast Boundary Fault System • Angelina-Caldwell Flexure • Sabine Uplift • Monroe Uplift	Low (R)	6(R) m = 3.5 to 4.1	m = 4.9 (R)
Gulf Coast Region	• South Louisiana Embayment • South Texas Embayment • Houston Embayment	Low (R)	4 (R) m = 4.2 to 4.4	m = 4.9 (R)
Central Texas Region	• Rio Grande Rift • Mexia-Talco Boundary Fault Zone • Llano Uplift • Permian Basin	Low (R) Surface faults considered inactive by LES	8 (R) m = 4 to 4.6	m = 5.7 (R)
Ouachita Region	• Surface and subsurface faults (R); earthquakes have not been directly related to these faults (LES) • Earthquakes widely distributed	Moderate (R)	m up to 4.8 (38 earthquakes m = 4.0 to 5.5 in Ouachita and Wichita Regions from 1894 to 1975) (a)	m = 5.7 (R)
Wichita-Arbuckle	• Meers Fault • Wichita Uplift • Arbuckle Mountains • Muenster Arch	Moderate (R)	m = 6.1 to 6.6 (S)[b] [inferred from Quaternary displacement along Meer's Fault (LES)]	6.8 (R)
Reelfoot Rift	----	Moderate (R)	4 (R) m = 4.5 to 4.9 R	6.8 (R)[c] (Consistent with upper limit assigned to Reelfoot Rift in general but "postulated" faults are assigned 7.4)
New Madrid Fault Zone	----	Moderate (R)	Many R m up to 7.4[d] indicates a magnitude greater than 8.0 is statistically likely every 550-1,200 years)[d]	m = 7.4 (R) (Nuttli and Herrmann, 1978, p. 79 (indicates the zone has an m_b = 7.5 recurrence in 800 years)[a]

Table 2.12 Summary of seismic activity related to geologic structures (continued)

Region	Structure	Seismic Activity (R) Region (S) Structure	Number of Earthquakes (R) Region (S) Structure	Maximum Magnitude Assigned (R) Region (S) Structure
Central Stable Region	• Anadarko Basin • Southern portion of Nemaha Ridge	Low (R)	4 (R) m = 4.4 to 5.5 (Nuttli and Herrmann cite 13 earthquakes m greater than 4.4 in Nemaha Ridge since 1867 to 1975)[a]	m = 5.7 (R)
Mississippi Embayment		Moderate (R)	13 (R) m = 4.5 to 5.4	m = 5.7

[a] The NRC staff's review of literature indicates that strong evidence exists for a magnitude 7+ earthquake within the past 1,000 to 1,400 years (Luza, et al., 1987; Ramelli, et al., 1987) although it has been relatively aseismic in historic past. The Meers Fault has a prominent scarp.

[b] Nuttli, O.W., and R.B. Herrmann, 1978.

[c] Law Engineering Testing Company, 1986.

[d] Johnston, A.C., 1982.

2.5.2.2.3 Design Earthquakes and Maximum Potential Earthquakes

The design basis earthquake (design earthquake) is assigned a return period of 500 years (NRC, 1988a). The applicant assigned values to three design earthquakes:

- Near-field, $m_b = 4.3$, located at basement depth of 5 kilometers (3 miles) and a distance of 14 kilometers (9 miles) from the site, with a peak horizontal acceleration at seismic basement at the site of 0.045g. Seismic basement is defined as rock having a shear wave velocity greater than 762 m/s (2,500 f/s)

- Mid-field, $m_b = 5.7$, located in the Ouachita Region at 105 kilometers (65 miles) from the site, with a peak horizontal acceleration of 0.04g at seismic basement at the site

- Far-field, $m_b = 6.7$, located in the New Madrid Fault Zone at 365 kilometers (225 miles) from the site, with a peak horizontal acceleration of 0.022g at seismic basement at the site.

The site is located in the near-field Interior Salt Basin Region. The applicant assigns a 4.9 maximum magnitude for this region in which six earthquakes between 3.5 and 4.1 have occurred.

The applicant assigns a 5.7 mid-field maximum magnitude design earthquake.

The applicant assigns a 6.7 far-field maximum magnitude design earthquake. The NRC staff notes that the maximum recorded earthquake was the magnitude 7.4 earthquake in 1811. In addition, there is evidence that the Meers Fault experienced a magnitude 7+ earthquake within the past 1,000 to 1,400 years (Luza, et al., 1987; Ramelli, et al., 1987). The Meers Fault is very close to the edge of the 320-kilometer (200-mile) radius from the site.

The maximum design earthquakes and potential earthquakes are summarized in Table 2.13.

2.5.2.2.4 Ground Motion and Response Spectra

Ground motion and horizontal and vertical ground response spectra were derived from earthquake data. The seismic parameters for the near-field, mid-field, and far-field design earthquakes were attenuated to the seismic basement at the site, and ground response spectra were developed for each of these three earthquakes.

Table 2.13 Maximum earthquake potential and design earthquakes

Field	Seismotectonic Region	Distance from Site (km)	Maximum Magnitude Assigned by Law Engineering Services	Design Earthquake	Maximum Peak Horizontal Acceleration at Seismic Basement at Site Based on Design Earthquake
Near-Field	Interior Salt Basin	14	5.7	4.3	0.045g
Mid-Field	Ouachita	105	5.7	5.7	0.04g
Far-Field	New Madrid Fault Zone	365	7.4	6.7	0.022g

Examples of recorded ground shaking from earthquakes (time histories) were used to develop the response spectra. Because of the relatively few earthquakes recorded in the regions, the applicant used some time histories from Canada (New Brunswick, m = 5.7 for the near-field event; Saguenay, Quebec, m = 5.9 for the mid-field event). For the far-field earthquake, it used an artificial earthquake of m = 6.7 because an earthquake of this magnitude has not been recorded in the eastern United States. This artificial earthquake used RVT to compute the peak acceleration. The response spectra were damped at 0.2, 0.5, 2.0, 2.5, and 10 percent. The spectra for each of the design earthquakes represent actual earthquakes and not an envelope or composite spectra. A comparison of these spectra shows how the near-field, mid-field, and far-field spectra differ. As expected, the more distant earthquakes have lower frequency content (longer periods) and larger values of acceleration and velocity. The values of displacement for the three spectra tend to cluster near 0.13 centimeters (0.05 inch).

The NRC staff notes that the use of data from the two earthquakes from Canada may be questionable. Their origins are in seismotectonic regions different from those which they are used to represent; thus, the differences may make their use invalid. The focal mechanism of these earthquakes may not be the same as those from the near-field and mid-field seismotectonic regions, and, thus, their seismic parameters may be different. A search for seismic records from earthquakes that occurred in the near-field and far-field should be made and compared to the Canadian records for similarities and dissimilarities. The NRC staff also notes that earthquake records are available from events in the New Madrid Fault Zone: June 13, 1975, m_b = 4 to 4.5; March 25, 1976, m_b = 5.0; March 25, 1976, m_b = 4.5 (aftershock) (Herrmann, 1977). These data could be scaled upward to the m_b = 6.7 used by the applicant to assess similarities and dissimilarities with the artificial, random vibration theory earthquake used in the analysis.

2.5.3 NRC Staff Findings and Evaluation

In general, the applicant's reports provide applicable geologic, geotechnical, and seismic data which can be used for evaluating the siting of the proposed uranium enrichment facility at the proposed site. This section summarizes the NRC staff's findings on the applicant data and analyses. In some instances, the data and analysis go beyond that needed for evaluation against the ANPR requirements. This section presents the NRC staff's comments on all applicant analyses.

The data used for the regional geology were based on literature searches; field work to verify reported active or suspected active faults was not done. The data used for the site geology were based primarily on information from borings and test pits from the present and previous investigations. The water table at the site is shallow, between 3 and 15 meters (10 and 50 feet) below the surface; perched water is also present.

The data used for the geotechnical analyses came from field samples and laboratory soils tests. The samples and the soils tests have followed ASTM standards. These data and their interpretations were used to make recommendations for site preparation and foundation characteristics for the facility. The data indicate that two stratigraphic layers have a low potential for liquefaction if the site is subjected to earthquake ground shaking of sufficient frequency, strength, and duration. These two layers are locally below the water table.

Geophysical surveys at the site consisted of two down-hole and two cross-hole surveys. Analysis of the compressional and shear waves indicate unusually high values (calculated) of Poisson's ratio, 0.45 to 0.48 (the theoretical maximum value is 0.50). These values were reported at levels both above and below the water table. Typical values for soils similar to those at the site are characteristically between 0.25 and 0.33 above the water table. The velocities from below the water table are affected by the water and should not be used to calculate Poisson's ratio. The high values of the ratio may result from the compressional and shear waves being affected by the grout at the boreholes. Another possibility may be related to misidentifying the first arrivals of seismic waves. In any case, the NRC staff recommends that the values of Poisson's ratio calculated from these surveys not be used in future refinements of the design.

The seismicity data used for the analysis were based on a literature search. Recurrence rates were not completely consistent with reports elsewhere in the literature. The design earthquakes are reasonable, and site-specific ground response spectra were prepared for near-field, mid-field, and far-field earthquakes. The design earthquakes used in the near-field and mid-field were derived from two different earthquakes in Canada. The NRC staff notes that these earthquakes may not be representative of earthquakes from the seismotectonic regions in the area near the site. The Canadian earthquakes may have different focal mechanisms and different frequency content than those occurring near the site. The far-field design earthquake from the New Madrid Fault Zone was artificial, generated by using RVT. This approach is generally accepted in areas where earthquake records are sparse. However, some time history records with magnitudes 4 to 5 are available from the New Madrid Fault Zone. These records could be scaled upward and compared with the one generated by RVT. If differences exist, then a decision must be made regarding which is the more reliable for design.

The applicant presented data and analyses which describe the geology and seismicity of the proposed CEC site and characterize the vibratory motion which might be experienced at the site during an earthquake. The NRC staff finds that the applicant's method for characterization of seismic risk, including identification of seismic zones, analysis of historical earthquakes and related faulting, and attenuation of earthquake effects from the source to the site, is a state-of-the-art method and is acceptable. The NRC staff reviewed the NEHRP seismic risk maps and determined that the peak horizontal acceleration predicted by the NEHRP method is less than or equal to the peak horizontal acceleration (i.e., $a_h = 0.046$ g) predicted by the applicant's method and is therefore acceptable.

The NRC staff reviewed both the horizontal and vertical response spectra proposed by the applicant. The applicant has selected as a basis for the analysis specific earthquakes which may not be fully representative of the local conditions but which are within the range of accepted professional judgment and are thus acceptable. For development of the horizontal response spectra, the applicant uses data and procedures acceptable to the NRC staff and the NRC staff concludes that the results are acceptable. For development of the vertical response spectra, the applicant uses an acceptable procedure in conjunction with Poisson's ratio data which the NRC staff believes may be high. The NRC staff concludes that the results for the vertical response spectra may not be conservative, and, thus has not relied upon this vertical response spectra in the review. The NRC staff followed the guidance of Regulatory Guide 1.60 in the review of structures, systems, and components.

The NRC staff concludes that the applicant's analysis of geology and seismology, in conjunction with the Southern Standard Building Code adopted by the applicant, and Regulatory Guide 1.60 procedures recommended by the NRC provides an acceptable basis for safety review of CEC structures, systems, and components.

3 PLANT DESIGN

The Claiborne Enrichment Center (CEC) is a process plant designed to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream enriched in the uranium-235 (U-235) isotope and a tails stream depleted in the U-235 isotope. The process, entirely physical in nature, takes advantage of the tendency of materials of differing density to segregate in the force field produced by a centrifuge. The chemical form of the working material of the plant, uranium hexafluoride (UF_6), does not require chemical transformations at any stage of the process. In the three primary steps of the process, UF_6 is volatilized from a feed cylinder, passed through the separative centrifuges, and condensed in product or tails cylinders. A block diagram of all steps in the process is presented in Figure 3.1. The process comprises a linear sequence of steps and does not recycle material from later steps in the sequence to earlier steps. The processing steps occur in seven buildings located within a 28 hectare (70 acre) controlled area. The layout of the controlled area is depicted in Figure 2.3, and the floor layout of the main building, the Separations Building, is presented in Figure 3.2. The description of plant design and operations presented in this chapter follows the flow of UF_6 from its reception at the site through processing to the disposition of product and tails material. Each sub-section first describes the facilities and equipment used in that step of the process and then describes the operations performed with the material and equipment. The primary process systems are described in this chapter, and auxiliary or support systems are described in Chapter 6. The descriptions are drawn from the CEC Environmental Report (ER) (LES, 1993b), Safety Analysis Report (SAR) (LES, 1992a), and selected Applicant Responses to NRC Requests for Additional Information (RAI) (LES, 1992a, 1992b, and 1992c). Detailed diagrams of CEC buildings and equipment are provided in the CEC SAR.

3.1 Feed Receiving and Storage

Facilities

CEC feed material in the form of solid UF_6 in cylinders is transported to the site on specially designed and fitted flat-bed trucks. Onsite, the feed material is delivered to the Cylinder Receipt and Dispatch Building (CRDB) for cylinder inspection, weighing, and testing. The CRDB is a steel-frame building with insulated sheet-metal walls and a reinforced concrete floor. The rectangular building is 30 meters (100 feet) long, 15 meters (50 feet) wide, and 10 meters (30 feet) high. An access corridor runs along the center of the long axis of the building, and roll-up doors on opposite sides of the building allow trucks delivering or removing cylinders to drive through. A 23-tonne (25-ton) capacity overhead bridge crane travels on rails mounted on opposite sides of the long axis of the building. The crane can move across the bridge and thus can cover almost all the floor area of the CRDB. An 18-tonne (20-ton) weighscale is located adjacent to the truck bay area. The interior floor area of the building stores feed and product cylinders on specially designed cradles. Storage space for 20 cylinders is provided in the CRDB. An outside storage area holding hardstands for 187 cylinders covers approximately one-half hectare adjacent to the CRDB.

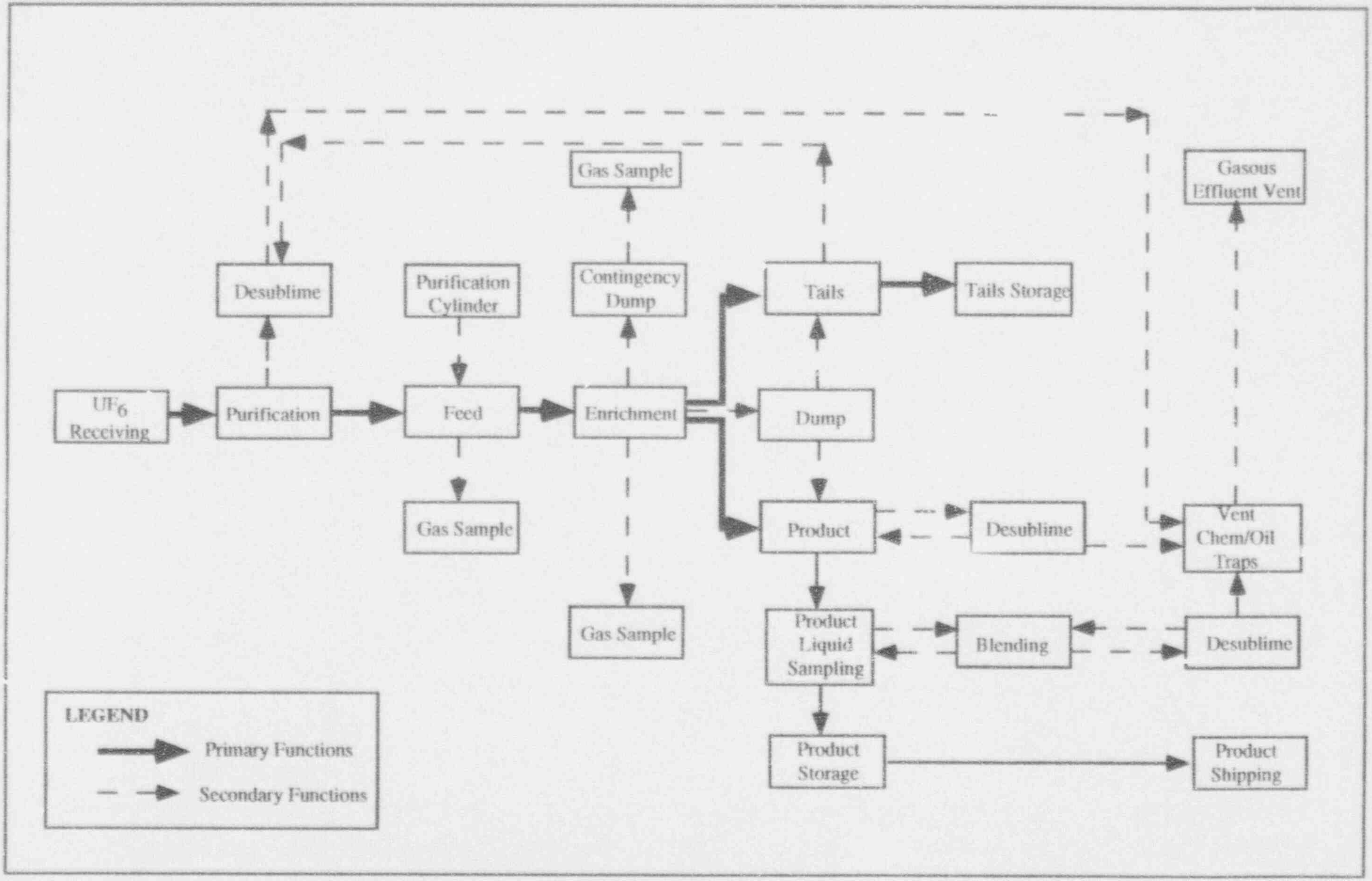

Figure 3.1 Block diagram of CEC processes

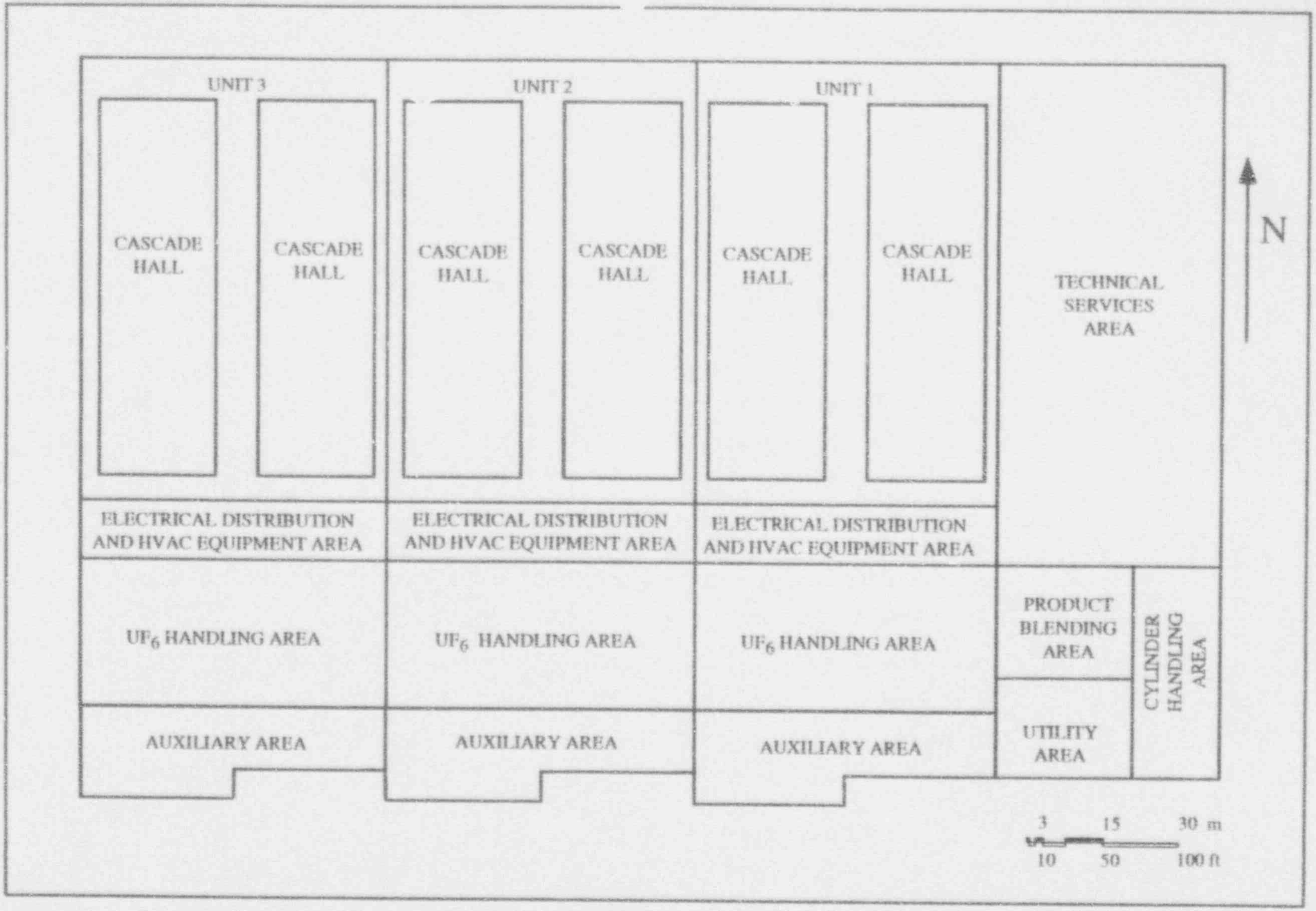

Figure 3.2 Floor plan of CEC Separations Building

Operations

An overhead crane unloads feed cylinders delivered to the CRDB. The cylinders are inspected for damage and general condition and weighed on the CRDB weighscale. The cylinders are then transported to the outside area for storage or to the Separations Building for processing.

3.2 Material Handling and Transfer

Facilities and Operations

Cylinders containing solid UF_6 feed, product, and tails material are stored outside the Separations Building and must be transported to and from storage areas and transferred to various process areas within the Separations Building. Mobile transporters, cranes, and rail carriages move these cylinders. The Separations Building is of pre-cast/pre-stressed concrete construction, 238 meters (780 feet) long and 140 meters (460 feet) wide. The floor area at ground level is 33,320 square meters (358,700 square feet).

Straddle carriers and modified forklifts transport cylinders from storage areas to and from the Separations Building. A straddle carrier is a wheeled vehicle fitted with a claw-like lifting mechanism which is hooked around the stiffening ring lifting lugs of a cylinder. The carrier is positioned over the cylinder and aligned with the long axis of the cylinder. A modified fork-lift is fitted with a forward-lifting mechanism which also hooks around the lifting lugs of a cylinder. The long axis of the cylinder is aligned perpendicular to the axis of the fork-lift. The lifting capacity of each vehicle is 20 tonnes (22 tons).

An overhead bridge crane delivers feed cylinders to the Cylinder Handling Area of the Separations Building. A diagram of the layout of this area is presented in Figure 3.2. The crane travels on rails parallel to the north-south directions and moves across the bridge in the east-west directions. The crane has a 25-tonne (28-ton) capacity and is used to unload cylinders from straddle carriers and forklifts, move cylinders to the weighscale, or load cylinders onto the rail transporter.

A rail system is embedded in the floor along the long axis of the Separations Building parallel to the rows of feed, blending, sampling, and take-off stations. A transporter mounted on these rails is used to move feed, product, and tails cylinders to and from these stations. Each cylinder is carried on a carriage which moves on a second set of rails installed on the rail transporter structure. The transporter is battery-powered and electric-motor driven with a maximum speed of 0.61 m/s (2 ft/s). The transporter is fitted with a rail bridge and hoist system in order to move the secondary carriage and cylinders into and out of stations. The hoist system is capable of lifting all types of cylinders over a range of 5 centimeters (2 inches) and matches rail systems installed in feed, sampling, blending, and take-off stations.

3.3 Feed Purification and Feed

Facilities

The function of the Feed Purification and Feed System is to remove light gas contaminants from the feed UF_6 and to provide a continuous, controlled flow of UF_6 to the separation cascades. The primary pieces of equipment for this purpose are the feed autoclave, feed purification desublimer and traps, feed purification cylinder station, and associated piping, valves, and controls. The configuration of equipment is represented in Figure 5.1. The CEC comprises three plant units each of which has independent, feed, enrichment, and product and tails take-off equipment. Each plant unit includes a single feed purification system and four feed autoclaves which are connected to a common plant unit header. The autoclaves are rotated through purification, heat-up, feed, and cool-down cycles to maintain desired feed to the cascades.

The feed autoclave is a horizontally mounted, carbon-steel, cylindrically shaped vessel with elliptical ends which is approximately 6 meters (20 feet) in length and 2.2 meters (7 feet) in diameter. The head at one end of the autoclave is hinged at the top to form a door which is operated by a hydraulic opening mechanism. Rails mounted on the floor support the cylinder carriage which can be rolled into and out of the autoclave. The cylinder cradle arrangement includes weight sensors for continuous monitoring of cylinder contents. The interior of the autoclave is heated by three separate resistance heaters and a two-speed fan with total heat transfer rate of 18,000 joules per second (J/s) (17 BTU/s). The autoclave air space communicates with the environment through two routes. The first piping line is a vent to the Separations Building air space which includes a normally closed shut-off valve. The second piping line connects the autoclave air space to the Gaseous Effluent Vent System (GEVS) through a hand-operated valve which has a mechanical interlock to the autoclave door. The autoclave also has interior piping and associated valves which provide the flow path for UF_6 being transferred to the desublimer or to the cascades. Inside the autoclave, the transfer piping has one hand-operated valve for line pump-down, one shut-off valve for cylinder/autoclave isolation, and one modulating valve for UF_6 flow control. In addition, a cylinder is fitted with an open/close valve which is connected to the autoclave interior piping through a flexible pipe called a pigtail.

The autoclave exit line connecting the feed cylinder to the feed desublimer has two open/close valves which direct flow to the desublimer or to the cascade header. Desublimation at ambient temperature is possible at pressures used in the purification system. Valves are in hot boxes, and lines are trace heated to prevent desublimation. The feed purification desublimer comprises four stainless steel tubes sealed inside a stainless steel cabinet. Each pipe is wrapped in a copper line capable of carrying hot refrigerant and a copper line capable of carrying cold refrigerant. The cabinet is filled with insulation and blanketed with nitrogen during operation to exclude moisture. Each desublimer tube is 0.41 meters (16 inches) in diameter and 6 meters (17 feet) in length with a volume of 0.75 cubic meters (24 cubic feet). The desublimer is connected to the GEVS through an exit line containing two valves, a pump, and two UF_6/hydrogen fluoride (HF) adsorption traps.

The feed purification desublimer is connected to a UF_6 cylinder contained in the feed purification cylinder station by a pipe containing three open/close valves, a short length of flexible pipe, and a cylinder valve. The cylinder station is a rectangular-shaped cabinet with cooling water connections. Cooling water from the main plant cooling water system is sprayed onto cylinders in the station and returned to the system through drain pipes located at the bottom of the cylinder station. Cylinders are placed inside the station on rail-mounted carriages with weighscale mechanisms for monitoring of cylinder contents.

Operations

Feed purification operations begin with loading of a full feed cylinder into a feed autoclave. The cylinder valve is connected to the autoclave interior piping through a flexible pipe and purification is accomplished in two stages. In the first stage, termed cold purification, the unheated cylinder is vented repeatedly in a batch fashion to a chilled desublimer in order to remove light gases present in the cylinder vapor space. The UF_6 transferred to the desublimer is solidified in the desublimer tubes at the desublimer operating temperature. Routing of the UF_6 flow is determined by valve positions selected on control switches for the autoclave and desublimer stations. The autoclave door is left open for this operation in order to automatically disable heater operation. Venting, including opening and closing of appropriate valves, is repeated until desublimer pressure measurement indicates that light gases have been effectively removed from the feed cylinder. Approximately 10 kilograms (22 pounds) of UF_6 are carried over to the desublimer in each of the 10 to 12 cold vent steps. After each step of the process, the vapor space of the chilled desublimer is vented to the GEVS through the feed purification system exit gas traps. Light gases, including air and HF, and a quantity of UF_6 determined by the vapor pressure of UF_6 at the desublimer temperature are released in this operation. At the completion of cold venting, the autoclave door is closed, and heating of the cylinder is initiated. Heater controls are interlocked to door and valve positions and to autoclave air space temperature and pressure in order to avoid overheating of the cylinder. When the UF_6 is fully liquified, the cylinder is again vented to the desublimer in order to remove light gas contaminants which may have been trapped in the solid UF_6. Generally only a single hot vent step is needed, and 1 kilogram (2.2 pounds) of UF_6 is transferred to the desublimer. At the completion of hot purification, the cylinder is ready for transfer of UF_6 to the cascades. An operational fill limit of 2,000 kg (4,400 pounds) is established for a four-tube desublimer but the solidified UF_6 is transferred to a purification cylinder when the desublimer contents reach an administrative limit of 400 kilograms (880 pounds) of UF_6.

Transfer of UF_6 from a hot, purified cylinder to the cascade header is effected through selection of valve positions at the autoclave state switch. Exit line pressures are monitored and control valve position is adjusted to maintain constant UF_6 mass flow rate. When the flow can no longer be controlled at the desired rate, the cylinder is taken off-line and replaced by another feed autoclave. The cylinder heel is transferred to the purification cylinder, and the feed cylinder is cooled and removed from the feed autoclave.

3.4 Enrichment

Facilities

The CEC enrichment system separates a stream of gaseous UF_6 produced in the feed system into a product stream enriched in U-235 and a tails stream depleted in U-235. The separation is effected in Urenco model TC-12 centrifuges which are grouped into arrays called cascades. The cascades in each of the three CEC plant units are organized into two assay units, each of which is comprised of seven cascades. Therefore, each plant unit has 14 cascades, and the CEC as a whole has 42 cascades. Each cascade is housed within a separate enclosure and comprises approximately 1,000 centrifuges. Each cascade in a plant unit is connected to an assay unit header which receives UF_6 feed from the plant unit header and feed autoclaves.

Equipment comprising the enrichment section includes the centrifuges, control valves, instrumentation, and associated piping. The centrifuge is a thin-walled, vertical, cylindrically shaped rotor which spins around a central post within an outer casing. The rotor is fabricated of carbon-reinforced epoxy, and the casing is aluminum. The centrifuge housing is installed upon a specially designed and leveled floorplate called a flomel. The space between the rotor and casing is maintained under vacuum to reduce drag. The rotor is driven by an electromagnetic motor which draws power from a run convertor at a frequency equivalent to the rotor speed. Under normal operating conditions, each cascade uses energy at a rate of 86 kilowatts. Each cascade has a closed-loop cooling water system to remove the heat generated by frictional losses and the electromagnetic motors. Cooling coils located at the top and bottom of the rotor remove heat and provide a temperature gradient which plays a role along with centrifugal force in producing the isotopic separation. Feed, product, and tails streams enter and leave the centrifuge through the central post. Enriched gas is withdrawn at the top of a rotor and depleted gas is withdrawn at the bottom of the rotor. Piping complexity is reduced by grouping the centrifuges into blocks which are connected in series and parallel fashion to constitute the cascade. Each centrifuge has an exit safety valve which is closed by excess pressure within the centrifuge. Each cascade receives UF_6 from the assay unit header through a control valve, resistor orifice, and controller which establish flow rate on the basis of monitored pressure level. Product flow is controlled in a similar manner.

Operations

Under normal conditions, the centrifuges operate continuously with minimal operator intervention. Automatic monitoring of centrifuge pressure, rotational frequency, and cooling conditions is continuous as described in SER Chapter 5. The pressure within each centrifuge is subatmospheric, and each centrifuge contains 10 grams (0.02 pounds) of UF_6. During normal operation, mobile pump sets are used to draw samples periodically from each cascade. The samples are desublimed into flasks by using liquid nitrogen, and non-condensible gases are vented to the GEVS. Ingress of light gas or process upsets can cause destabilization of the centrifuge rotor and a resulting failure, termed a "crash". In a centrifuge failure, rotational energy is converted to heat, the rotor disintegrates, and a quantity of gas is generated in the disintegration process and subsequent reaction with UF_6. A pressure pulse occurring during the crash closes isolation valves and separates the failed centrifuge from the balance of the cascade. Solid reaction products accumulate in the bottom of the failed centrifuge, and over a period of weeks, the reaction gases leak into the cascade header and are removed through the GEVS. The failed centrifuge remains in place but no longer contributes to the separation capacity of the cascade.

3.5 Product and Tails Take-off

The primary function of the product and tails take-off systems is to compress streams of UF_6 from cascade pressure to an elevated but sub-atmospheric pressure to desublime UF_6 in product or tails cylinders. A secondary function of the product and tails take-off systems is to provide for rapid removal, or dumping, of the inventory of a cascade to product or tails cylinders. The equipment used for the product and tails take-off is similar but not identical. Flow schematics of the product and tails take-off pumps and cylinder stations are presented in Figures 5.2 and 5.3, and Figures 5.4 and 5.5, respectively.

Product Take-Off Facilities

Enriched UF_6 is compressed to desublimation pressure in a two-step manner. In the first step, the product stream from each cascade is passed through two series, low-pressure, vacuum pumps as pressure is increased from 275 to 4,830 pascals (0.04 to 0.7 psia). The seven streams from each assay unit cascade are then combined and passed through two parallel, high-pressure vacuum pumps. Pressure is increased from 4,830 to 43,990 pascals (0.7 to 6.4 psia). Desublimation of UF_6 is possible at ambient temperature at process pressure downstream of the high-pressure vacuum pumps. Hot boxes for pumps and valves, and line heat tracing are used to prevent desublimation. Flows from the two assay units comprising a plant unit are routed to one of the ten product take-off stations serving the plant unit. The take-off station is a rectangular box 1.6 meters (5 feet) wide, 1.6 meters (5 feet) high, and 2.5 meters (8 feet) long. The box is double-walled, insulated, and fitted with rails which support the cylinder carriage. Load cells are provided, and the heat of desublimation is removed with cooling air routed through closed-circuit feed and return lines. A single-tube desublimer similar in design to the feed purification desublimer removes light gases. Three product take-off desublimers are provided for each plant unit. The stainless steel desublimer tube is housed in an insulated stainless steel cabinet and is wrapped with two copper tubes which carry hot and cold refrigerant. The desublimer exit line is valved and includes a vacuum pump and chemical traps for removing UF_6 and HF vented with the light gases.

Product Take-Off Operations

Flow from the cascades through the low- and high-pressure vacuum pumps is normally continuous and is monitored with pressure sensors. Controllers on pump intake and exhaust lines terminate flow if pressure is outside of prescribed operating ranges. During maintenance of a cascade low-pressure pump, flow is diverted to the low-pressure pumps of an adjoining cascade which are sized for this service. During maintenance of one of the two parallel, assay unit, high-pressure pumps, the remaining pump is sized to handle the full flow of the assay unit. Of the ten product take-off stations provided for a plant unit, three are normally on-line at any given time. Take-off station activities include loading of an empty product cylinder into the station and connection to the take-off header by using flexible pipe. The empty cylinder and process lines are evacuated to a pressure of one pascal and held at that pressure for 30 minutes to detect potential leakage. Cooling air flow is initiated and product is desublimed into the cylinder. Pressure in the cylinder rises due to the accumulation of light gas which preferentially flows to the product end of the cascade. When cylinder pressure reaches 4,480 pascals (0.65 psia), the inlet line valve is closed, and the cylinder is vented in batch fashion to the desublimer. In each vent step, the desublimer is itself vented to the GEVS and cylinder venting is repeated until the cylinder and desublimer pressures differ by less than 10 pascals (0.001 psia). One or two venting sequences are required for a product cylinder with less than 20 kilograms (44 pounds) of UF_6 carried over to the desublimer from each cylinder. When load cells indicate that a cylinder is full, inlet and cylinder valves are closed, the lines are evacuated, and the cylinder is disconnected and removed from the take-off station. When process records indicate that a desublimer tube contains 100 kilograms (220 pounds) of UF_6, the desublimer is heated, and the contents are transferred to a standby product cylinder station.

Tails Take-Off Facilities

Depleted UF_6 is removed from the enrichment system in a process similar to product removal. Tails from each cascade are compressed in two series, low-pressure vacuum pumps from cascade pressure to 4,830 pascals (0.7 psia). The combined flow from the seven cascades of an assay unit is compressed to 22,750 pascals (3.3 psia) in three parallel, high-pressure vacuum pumps. Desublimation is possible at ambient temperature at process pressure downstream from the high-pressure pumps. The high-pressure pumps and downstream valves are in hot boxes, and downstream lines are heat-traced to prevent desublimation. The tails stream from the high-pressure pumps is routed to ten tails take-off stations, which serve the plant unit. The take-off station is a rectangular, insulated box with capacity for spray water cooling of a tails cylinder. The box is 2 meters wide (6 feet), 3 meters (7 feet) high, and four meters (13 feet) long. The cylinder is contained within the station on a rail carriage similar to the product take-off station. The cooling system is closed-circuit, and the rail supports are fitted with load cells to monitor cylinder contents.

Tails Take-Off Operations

Tails take-off operations are similar to product take-off operations except that cylinder venting is infrequently required. Flow through the vacuum pumps is continuous and cross-piping and excess capacity are provided for maintenance and replacement. Empty cylinders are placed in take-off stations, connected to the piping, and evacuated as described for product cylinders. After successful leak testing, cooling water flow is initiated, the inlet valves are opened, and desublimation begins. Although venting is seldom required, the cylinder may be vented to the feed purification system if necessary. When the load cells indicate that the cylinder is full, the valves are closed, the piping evacuated and disconnected, and the cylinder removed from the station.

Product, Tails, and Contingency Dump

Extraordinary conditions may require the rapid removal of the inventory of a cascade, assay unit, or plant unit. In these circumstances, the inventory may be removed through the Tails, or Product Take-off Systems, or through a dedicated system, the Contingency Dump System. In the case of removal through either the Tails Take-off System or the Product Take-off System, pressure to the low-pressure vacuum pumps is increased, and mass flow to the cylinder stations is increased. The UF_6 inventory is then desublimed into product or tails cylinders. If the product and tails removal functions are unavailable, the plant inventory may be removed in a Contingency Dump System provided for each assay unit. The system is comprises sodium fluoride beds to adsorb UF_6 and surge vessels to provide flow control. Each Contingency Dump System has seven parallel adsorbers/surge vessels, which service the seven cascades of the assay unit. Light gases passing through the dump system are released through the GEVS to the atmosphere.

3.6 Product Sampling and Blending

The function of the product sampling system is to certify that product meets customer specifications. The function of the product blending system is to provide the ability to produce a range of compositions of enriched product with a minimum complexity of separation system configuration. Schematics of the blending and sampling systems are presented in Figures 5.7 and 5.6, respectively. Each plant unit has a sampling system comprised of two autoclaves, and all three plant units are served by a single blending system comprising two autoclaves and five blending cylinder stations.

Product Sampling Facilities

Sampling of product cylinders is accomplished in autoclaves specially designed for that purpose. The sampling autoclaves are horizontal, cylindrical, carbon-steel vessels approximately 1.6 meters (5 feet) in diameter and 4 meters (15 feet) in length. Rail structures which support cylinder carriages are used to move cylinders into and out of the autoclaves. Three electrical heaters and an air fan are used to liquify and homogenize UF_6 in the product cylinder. The autoclave air space is connected to the Separations Building air space and to the GEVS through valved piping. Sample manifolds are connected to the cylinder valve inside the autoclave, and the autoclave air space is connected to the GEVS through a open/close valve. The front support of the autoclave is hinged, and the rear support is mounted on a hydraulic lift, which can tilt the autoclave, cylinder, and sampling manifold. Autoclave air space temperature and pressure are monitored by redundant Class I systems, which shut down heater function if temperature or pressure limits are exceeded. Cooling of the autoclave is provided by a closed-circuit, non-contact cooling water system.

Product Sampling Operations

A rail transporter loads a full product cylinder into the sampling autoclave. The cylinder is then clamped in place to prevent movement. The sampling manifold is connected to the cylinder and a vacuum pump set evacuates the piping. Once leak tests are complete, the autoclave door is closed and the heating cycle begun. The UF_6 is liquified, allowed to mix convectively for 16 hours, and heated an additional 2° C (4 °F) to ensure sublimation of UF_6 in the sample manifold. The autoclave is then tilted 30 degrees, and liquid UF_6 pours into the sample bottles attached to the sample manifold. The heater and fan are then activated to heat the autoclave an additional 8° C (14 °F) to evaporate UF_6 from the sampling piping. The autoclave is returned to the horizontal position and cooled by using water circulated through the non-contact cooling coils. Once the autoclave is cool, its air space is checked for leaks and the cylinder removed by using the rail transporter.

Product Blending Facilities

Major pieces of equipment involved in a blending operation include two donor autoclaves, one blended product cylinder station, one product blending desublimer, and associated valves and piping. Each donor autoclave is a horizontal, cylindrical, carbon steel vessel 1.6 meters (5 feet) in diameter and 4 meters (13 feet) in length. The autoclave is fitted with rail supports for the cylinder carriage and has a hinged door at one end. The autoclave is indirectly heated by three resistance heaters and an air fan. Closed-circuit indirect water cooling coils remove heat. The autoclave air space temperature and pressure are monitored by Class I systems, and the air space is connected to the Separations Building atmosphere and to the GEVS through valved piping. The two autoclaves are connected by common piping to the blending receiver cylinder station and to the blending system desublimer. The cylinder station is a double-walled, insulated, rectangular, carbon-steel box 1.7 meters (5 feet) wide, 1.7 meters (5 feet) high, and 2.5 meters (8 feet) long. The station is cooled by a closed-circuit air system and fitted with load cells to monitor cylinder contents. The desublimer is a single stainless-steel tube wrapped in copper, hot and cold, refrigerant tubes inside an insulated cabinet. The desublimer tube exit line is connected by two valves, a vacuum pump, and adsorber traps to the GEVS.

Product Blending Operations

The rail transporter loads full product cylinders and an empty receiver cylinder into the donor autoclaves and receiver cylinder station, respectively. The cylinders are connected to the process lines, and all lines exposed to air are evacuated. Before heating, the pressures of the donor cylinders are measured, and the cylinders are vented to the desublimer if necessary. The donor cylinders are then heated and the UF_6 liquified. The specified amount of UF_6 is transferred from each donor cylinder to the receiver cylinder until the receiver cylinder is full or the donor cylinders are empty. At the completion of operations, isolation valves are closed, connector lines evacuated and the receiver cylinder removed from the station. Heels from empty donor cylinders are transferred to the desublimer and then to a standby receiver cylinder station. Full heels cylinders are used as blending stock or sold individually. Donor cylinder connector lines are evacuated and empty cylinders removed from the autoclave.

3.7 Product and Tails Storage

Facilities and Operations

Outside storage areas are provided for UF_6 feed, product, and tails cylinders. All UF_6 is stored in the solid state at ambient temperature with subatmospheric pressure. The Product and Feed Storage Area is located southeast of the Separations Building and covers 0.5 hectare (1.2 acres). Two Tails Storage Areas located southwest and southeast of the Separations Building cover a total area of 6.1 hectares (15.2 acres). The storage areas are paved and are sloped 0.2 percent from horizontal toward the southeast to drain rainwater. Cylinders are stored on concrete saddles approximately 20 centimeters (8 inches) above ground level. Cylinders are not stacked for storage and adequate clearance is provided for mobile carriers to access the yards.

3.8 Centrifuge Assembly

Facilities and Operations

The centrifuges used at the CEC are assembled in the Centrifuge Assembly Building (CAB) from components received from off-site. The building is a metal frame structure 42.7 meters (140 feet) wide, 85.3 meters (280 feet) long, and 12.2 meters (40 feet) high. Radioactive materials are not handled or stored in the CAB. Components are received in a storage area and unloaded from land/sea containers by using a 9-tonne (10-ton) overhead crane. Centrifuge components are assembled and tested in a clean work area separated from the storage area by air locks. Assembled centrifuges are transferred to the Separations Building through an airlock corridor between the two buildings.

3.9 Start-up

Start-up of the CEC involves purging of lines and equipment, calibration and testing of instruments, verification of flow configuration, run-up of centrifuges, and introduction of UF_6 feed material. The process equipment constructor verifies the as-built system against design, purges lines and equipment, completes hydrostatic and pneumatic testing of lines and equipment, and calibrates instruments before pre-operational testing. The pre-operational test phase comprises four steps. First, the flow path from the feed autoclaves to the product and tails take-off stations is verified. Second, the cascades are evacuated, run-up, and held at vacuum for 150 hours. Run-up converters start the centrifuges in a sequence of frequency steps over a period of several hours. Third, a small amount of feed UF_6 is introduced into the equipment and allowed to react with contaminant residues present in the equipment as a consequence of fabrication and installation. Fourth, design quantities of UF_6 are introduced and cascade conditions verified through sampling of cascade enrichment settings.

4 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structural, mechanical, and facilities design criteria developed for the Claiborne Enrichment Center (CEC) are the basis for evaluating the systems proposed for enriching uranium hexafluoride (UF_6), maintaining safe operating conditions, and protecting public health and safety. This chapter describes the principal design criteria, identifies structures, systems, and components important to safety; and evaluates the safety systems under design basis conditions. These evaluations are based on the plant design described in SER Chapter 3. The major structures and components are the Separations Building and the cylinders, autoclaves, and associated piping which contain UF_6. The first section summarizes the results; the remaining sections detail the evaluations. The NRC staff reviewed the applicant's analysis of structures, systems, and components important to safety and completed an independent analysis of this topic. The NRC's and applicant's analyses were consistent in identifying important-to-safety systems.

4.1 Conformance with the ANPR General Design Criteria

The NRC has developed general design criteria applicable to centrifuge enrichment of UF_6. The criteria are applied to the appropriate processes, with their implications for safety paramount. The Advance Notice of Proposed Rulemaking (ANPR) (NRC, 1988a) specifies that the design must:

- Protect against loss of confinement capability when such loss of capability results from any single failure of a system having safety significance
- Provide diversity in safety systems commensurate with their safety function
- Minimize non-random, concurrent failures of important elements in protection systems
- Provide criteria and bases for resistance of parts of the facility to upper limit accidents
- Provide employees adequate protection from hazards.

Both the NRC staff's and applicant's analyses identified feed, blending, and sampling autoclave heater protection controls as important to safety. Because failure of these protection controls could lead directly to an adverse safety condition, the controls are categorized as System Class I (Category A) and their Quality Assurance level is 1 (see Chapter 12). The applicant's safety analysis evaluates the function of the Separations Building, autoclaves, and System Class I components under design basis conditions and limits transporter fuel inventory in a proposed license condition. The NRC staff has reviewed the applicant's analyses of response to design basis events, performed supplementary calculations and concluded that the systems function under the design basis conditions. Consequently, the NRC staff and applicant arrive at a consistent set of controls through analysis of structures, systems, and components.

The heater protection systems are of a multiple redundant series design and thereby protect against concurrent failures and prevent loss of confinement in the event of a failure by a single component. The systems are also diverse in that the safety function is initiated by off-normal conditions of both temperature and pressure. The NRC staff concludes that the proposed design basis is adequate to ensure continued function of the systems under identified design basis conditions defined by upper-limit accidents initiated by winds and tornadoes, earthquakes, and floods. On the basis of a review of safety systems, including criticality control, fire protection, and ventilation, the NRC staff also concludes that the system designs provide adequate protection of workers and the public.

4.2 Classification of Structures, Systems, and Components

The design of the Louisiana Enrichment Services (LES) uranium enrichment process uses a large number of structures, systems, and components to concentrate the uranium-235 (U-235) isotope in the feed material, to maintain adequate plant environmental conditions, and to protect worker and public health and safety. Detailed evaluation of each of this large number of structures, systems, and components is not necessary because only a limited number play a significant role in protecting public health and safety. Section 4.2.1 specifies the criteria used to identify structures, systems, and components important to safety. Section 4.2.2 presents a summary description of applicant's analysis of structures, systems, and components important to safety. Section 4.2.3 presents the NRC staff's analysis of structures, systems, and components important to safety. The NRC staff's and applicant's analyses are consistent in identifying the same set of components important to safety.

4.2.1 Criteria for Safety Significance

The NRC has published an ANPR (NRC, 1988a) which provides guidance on UF_6 safety and on standards for chemical effects related to release of UF_6. Release of UF_6 poses a dual radiological and toxic chemical hazard because UF_6 released to the atmosphere reacts exothermally with water vapor to form uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF). Inhaled or ingested uranium can cause damage to the kidneys and, sometimes death, and hydrogen fluoride is a corrosive chemical which attacks all tissue and is also potentially life-threatening. The NRC evaluation (NRC, 1991a) of the acute radiological and toxic chemical effects resulting from releases of UF_6 concludes that, for acute exposures, the toxic chemical effects exceed the nonstochastic radiological effects. The evaluation also concluded that the chemical effects of an intake of about 10 milligrams of uranium in soluble form are comparable to the radiological effects of an acute whole-body dose of 25 rem, which would result from the intake of a much larger quantity of uranium. Both exposures are just below the threshold for clinically observable nonstochastic effects. Similarly, exposure to HF at a concentration of 25 milligrams per cubic meter for 30 minutes was identified as the level for no significant effects, either short-term or long-term. The threshold concentration upper limit for exposure to HF was found to be inversely proportional to the square root of exposure time. By Commission Order (NRC, 1991b) facilities designed to preclude events which cause chemical effects of these magnitudes should not pose a significant adverse threat to public health and safety. Therefore, the uranium intakes and HF concentration limits specified in NUREG-1391 are the criteria used to identify structures, systems, and components as important to safety.

The enrichment process uses a variety of chemicals and energy sources which, while not present in quantities comparable to UF_6, may pose a threat to public health and safety. To maintain a consistent approach for uranium releases, structures, systems, and components are identified as important to safety if their function is required to prevent exposure of the public to chemicals at a concentration just below that which causes clinically observable effects. For specific chemicals, Emergency Response Planning Guides or American Conference of Governmental Industrial Hygienist (ACGIH) Time Weighted Average (TWA) (ACGIH, 1986) concentration levels are adopted as the criteria for identification of structures, systems, and components as important to safety.

4.2.2 NRC Staff Review of Applicant's Analysis of Structures, Systems, and Components Important to Safety

Applicant's analysis of structures, systems, and components important to safety is based on the premise that containment of UF_6 is the primary CEC safety concern. A four-step procedure constitutes the analysis. First, the criteria for identifying structures, systems, and components important to safety are specified. The applicant adopted the NUREG-1391 (NRC, 1991a) intake of 10 milligrams of uranium by an off-site individual as the appropriate criterion as directed by the Commission Order. Second, an atmospheric dispersion and uranium intake analysis is performed to identify release quantities of UF_6 which would result in off-site uranium intakes of 10 milligrams. The applicant considered that releases may be either buoyant or non-buoyant and adopted the dispersion analysis used by the NRC in the evaluation of emergency preparedness requirements at fuel cycle facilities (NRC, 1988b). Third, plant equipment, inventories, and failure scenarios are evaluated for buoyant releases. Fourth, failure inventories and release scenarios are evaluated for potential non-buoyant releases.

For buoyant releases, applicant's dispersion/intake analysis identified 1,100 kilograms as the quantity of UF_6 which would result in a 10 milligram uranium intake off-site. The applicant then reviewed plant area inventories and concluded that only cylinders contain UF_6 quantities greater than 1,100 kilograms. The applicant proposed that rates of UF_6 sublimation are low enough that a release of 1,100 kilograms from the solid state is not feasible. The NRC staff's analysis supports this assumption. Therefore, cylinders containing liquid UF_6 are identified as the potential source of releases which could exceed the criteria. Independent autoclave air space temperature and pressure sensors with automatic heater shut-off capability are designated as Class I systems, which precludes the potential occurrence of this release scenario.

For non-buoyant releases, the applicant's intake and dispersion analysis considered flow through a Separations Building door and estimated that 119 kilograms of UF_6 is the limiting release quantity. The applicant reviewed plant area inventories and concluded that cylinders, desublimers, and cascade hall piping each contain more than 119 kilograms of UF_6. Release scenarios developed for these plant areas included pipe breaks and pump fume release events.

The release analysis considered air dilution and deposition of UO_2F_2 within the Separations Building. The analysis results indicated that off-site uranium intakes for all scenarios would be less than the NUREG-1391 criteria. The NRC staff concurs with the result of this analysis for Separations Building systems and components.

4.2.3 NRC Staff Independent Analysis of Structures, Systems, and Components Important to Safety

In order to provide a differing perspective and to ensure that all systems important to safety have been identified, the NRC staff completed an independent analysis of structures, systems, and components important to safety. CEC design and process descriptions used in the analysis are drawn from the CEC Safety Analysis Report (SAR) (LES, 1993a) and Environmental Report (ER) (LES, 1993b).

4.2.3.1 Evaluation Methods

Identification of structures, systems, and components important to safety is based on estimating hazardous chemical exposures and concentrations in the environment surrounding the site. This calculation requires identifying hazardous chemicals on the site, identifying the critical exposure location, considering release scenarios, and estimating exposure or concentration at the critical location. Completing these calculations for all combinations of chemicals and components is a lengthy process which is avoided through use of a screening procedure. The screening procedure, as shown in Figure 4.1, incorporates hazard audit, process characterization, dispersion analysis, and analysis of equipment, and instruments and controls. The screening analysis is conducted at two levels of detail which are differentiated primarily by the amount of information incorporated into description of the release scenario.

The first stage of the analysis uses the hazard audit and process information to identify the maximum quantities and flows of material at selected locations throughout the plant. Maximum instantaneous and continuous releases are postulated on the basis of this review, and a dispersion analysis estimates intakes and concentrations at the critical receptor location. Review of a set of potential release scenarios and of instrument and control functions supported selection of two hours as a conservative estimate of the duration of a release. The review concluded that multiple signals of off-normal conditions would be reported at Local Control Centers and at the Central Control Room. These signals would occur early in the release event and thereby allow operator response to terminate the flow. Minimal information on equipment function is used at this stage, and the release scenario is of a generic character.

Release scenarios included continuous and instantaneous releases both inside and outside the Separations Building. If the estimated doses and concentrations are less than the review criteria, then the structures, systems, and components used to contain the material at that plant location are not important to safety; if the exposures or concentrations exceed the criteria, the screening proceeds to the second stage.

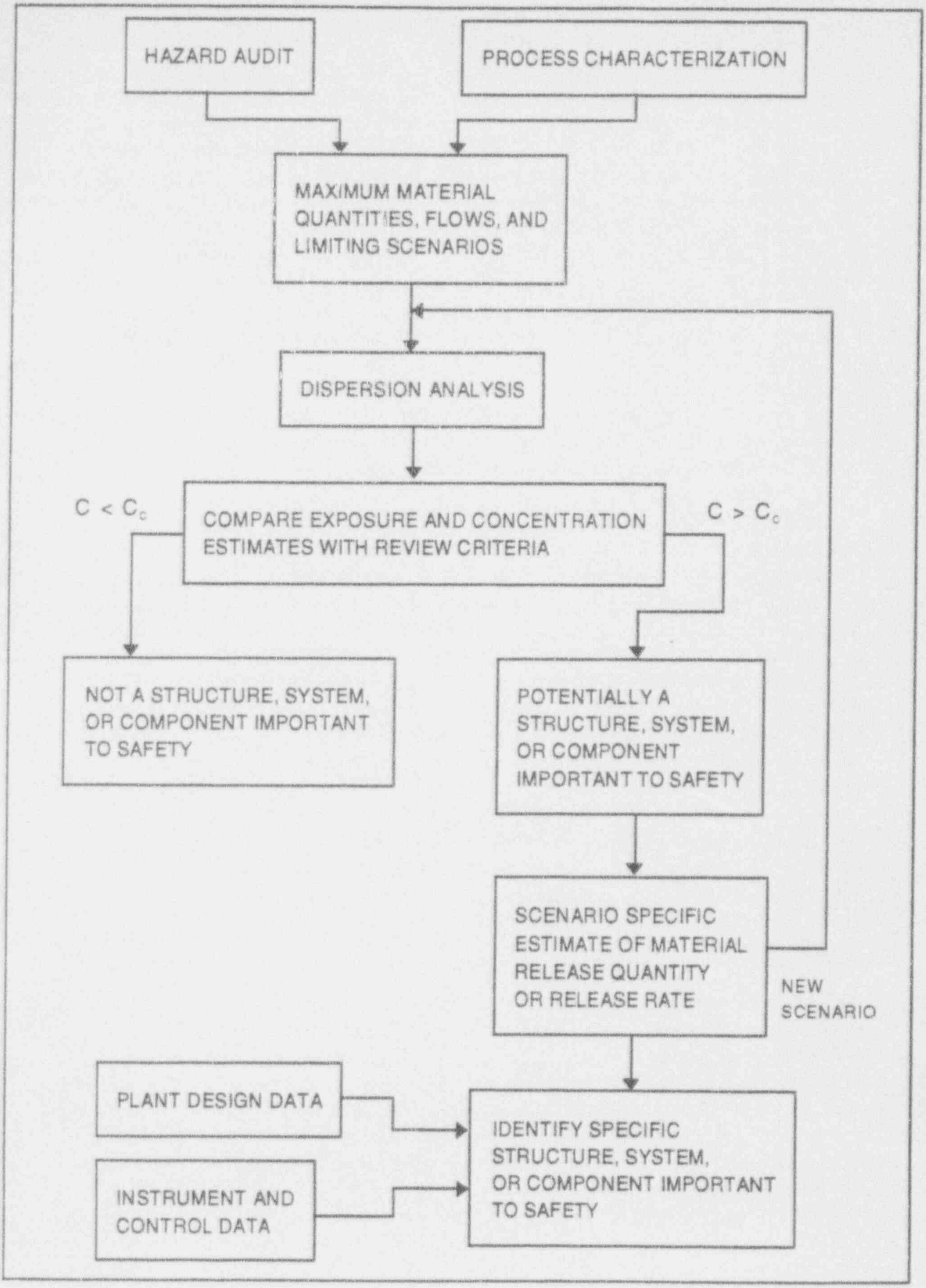

Figure 4.1 Safety review screening procedure

The second stage of the analysis uses the hazard audit, process characterization, and dispersion analysis of the first stage with more detailed specification of the initiating event and review of equipment function in order to develop more realistic estimates of the quantity or rate of release of material. The response of the structure, system, or component to the initiating event is considered at this stage of the analysis. If analysis demonstrates that the system does not fail if the initiating event occurs, then a release does not occur. The severity of the initiating event is limited to the design basis specified in the ANPR. A single active component failure criterion is applied in developing the scenarios considered in the screening procedure. Applying this principle allows identification of specific process conditions--for example, temperatures, pressures, and flows--which are the basis of the analysis of system response. The attention of individual operators to non-alarmed functions is not credited for preventing a hazardous condition. The attention of multiple operators over a period of time to non-alarmed functions is credited for preventing a hazardous condition. The response of a single operator to a single alarm is not credited for preventing a hazardous condition or for terminating a continuous release, but the response of more than one operator to more than one signal is credited for completion of such actions.

Estimating the source term at the second stage uses mass, momentum, and energy balance based physical models to characterize the release. Exposures and concentrations estimated in the dispersion analysis are again compared to the review criteria. If the exposures and concentrations are less than the review criteria, then the structures, systems, and components which contain the material at that plant location are not important to safety; if the exposures or concentrations exceed the review criteria, then at least some structures, systems, or components used to contain the material are important to safety. The details of the release scenario as well as the specific functions of equipment, and instruments and controls are reviewed to identify the specific structures, systems, or components which are needed to prevent the occurrence of the release. These specific structures, systems, or components are designated as important to safety.

The individual elements of the screening procedure use engineering analysis described in detail in other sections of this SER, and use dispersion analysis suggested in prior NRC guidance. These elements are described in the following paragraphs.

Hazard Audit

An audit conducted for the CEC identified large quantities of UF_6 as the primary hazard associated with operation of the facility. SER Chapter 11 presents a detailed description of the audit. The material is stored in the UF_6 Handling, Blending, and Cylinder Handling Areas within the Separations Building, and in the Tails Storage, the Product Storage, and Feed Storage Areas outside the Separations Building. Secondary quantities of UF_6 include the inventories of the piping and centrifuges used in the separation process. The inventory of UF_6 in each cascade is slightly less than 10 kilograms; because there are 42 cascades, the total inventory of UF_6 in the cascades is less than 500 kilograms. Thus, the inventory of UF_6 in the centrifuges and piping is of little importance to the safety of the system. Streams of

UF_6 moving through the CEC systems were also considered as potential hazard sources for continuous releases. A schematic of the flow configuration for one of the three plant units is presented in Figure 4.2. The flow configurations for the other two plant units are identical to the system represented in this figure. The maximum flow rate in any one pipe is approximately 50 grams per second, with product flows from individual cascades as low as 0.7 grams per second. The hazard audit did not identify quantities of hazardous materials other than UF_6 which could provide a basis for identifying a protective system as important to safety.

Dispersion Analysis and Release Scenarios

Dispersion analysis to identify systems important to safety considers release modes and scenario-specific factors in addition to the meteorological phenomena normally considered in atmospheric dispersion analysis. Release modes include continuous and instantaneous (puff) releases, and scenario-specific factors include heat generation and mixing before release to the atmosphere. Reaction of UF_6 with atmospheric water generates heat, which increases the effective release height and decreases concentrations at ground-level receptors. Each gram of UF_6 contains 0.68 grams of uranium. Evaluating the magnitude of this effect depends in a complex manner on reaction kinetics and the rate of entrainment of air into the UF_6 plume. To provide a simplified, conservative analysis, buoyant plume rise is not considered in the screening level analysis. The breathing rate used in the analysis, 3.47×10^{-4} m^3/s, is recommended by the NRC (NRC, 1974) for estimation of intakes in early stages of hypothetical accidents.

UF_6 releases inside the building are diluted before their release to the atmosphere. Evaluation of the degree of mixing depends in a complex manner on release dynamics and building air flow patterns. To provide a conservative analysis, the release scenarios considered the function of the building ventilation system. In the first scenario, the ventilation system remains functional, mixing in the building air is not considered, and the material is released directly to the stack without dilution in the building air. In the second scenario, the building ventilation system is shut down in contaminated areas, and the released material is dispersed into the accident compartment. In the absence of forced ventilation, normal wind flow outside the building develops a draft through the building; the result is a leakpath, or continuous release, represented as occurring at ground level. The features of the Separations Building which are relevant to this release estimate are represented in Figure 4.3. Wind flowing around the building induces a pressure drop along the path of the access corridor. Flow through this leakpath is limited by the resistance to flow through clearances around each of the doors along the access corridor. The magnitude of this resistance is estimated by using standard correlations (Blevins, 1984). Using wind speeds occurring less than 5 percent of the time (approximately 5 m/s), the NRC staff developed an estimate of leakage flow of 0.039 m^3/s (80 cfm) along the access corridor. Two doors in series separate each UF_6 Handling Area from the access corridor. Using the pressure distribution estimated for the

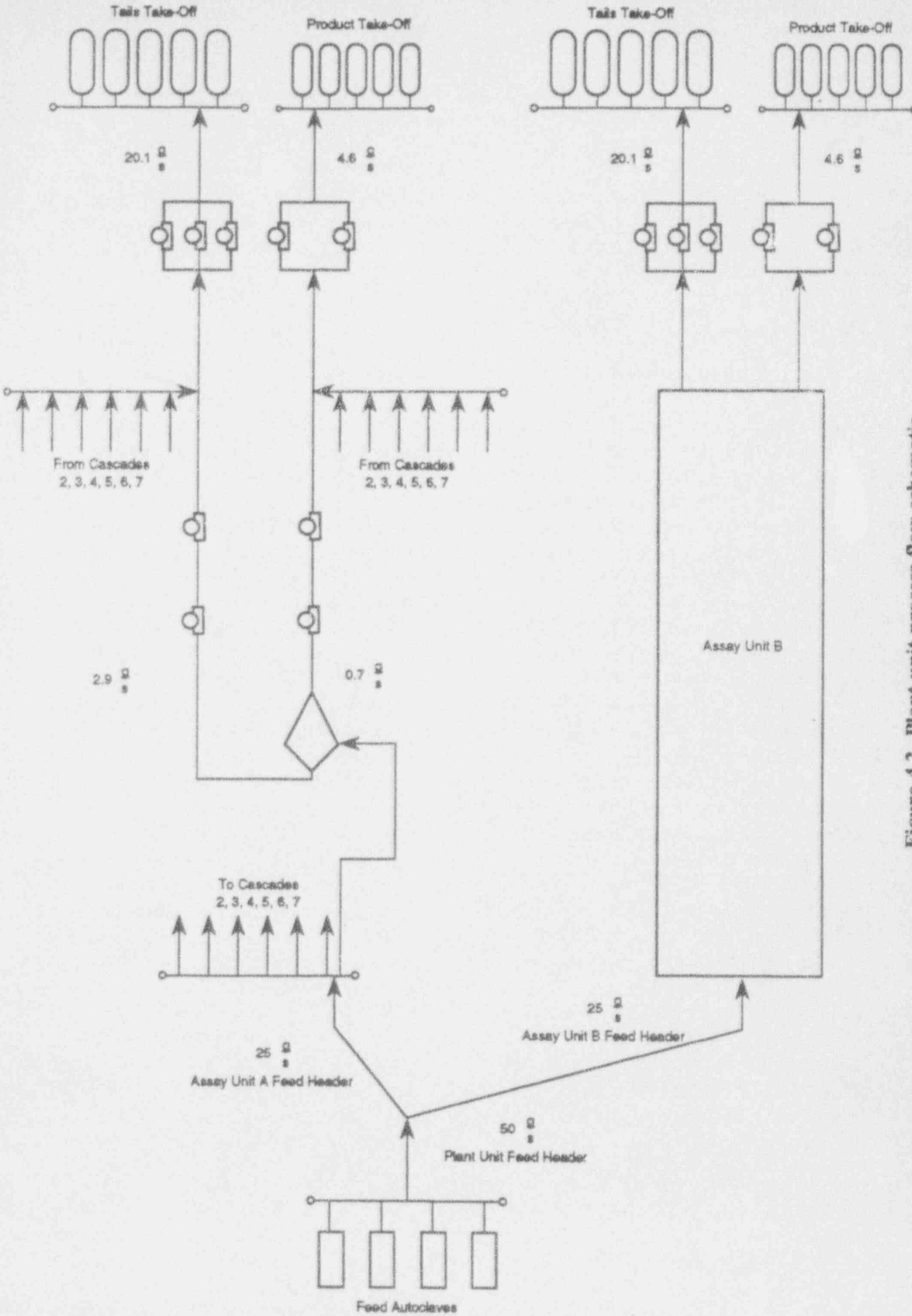

Figure 4.2 Plant unit process flow schematic

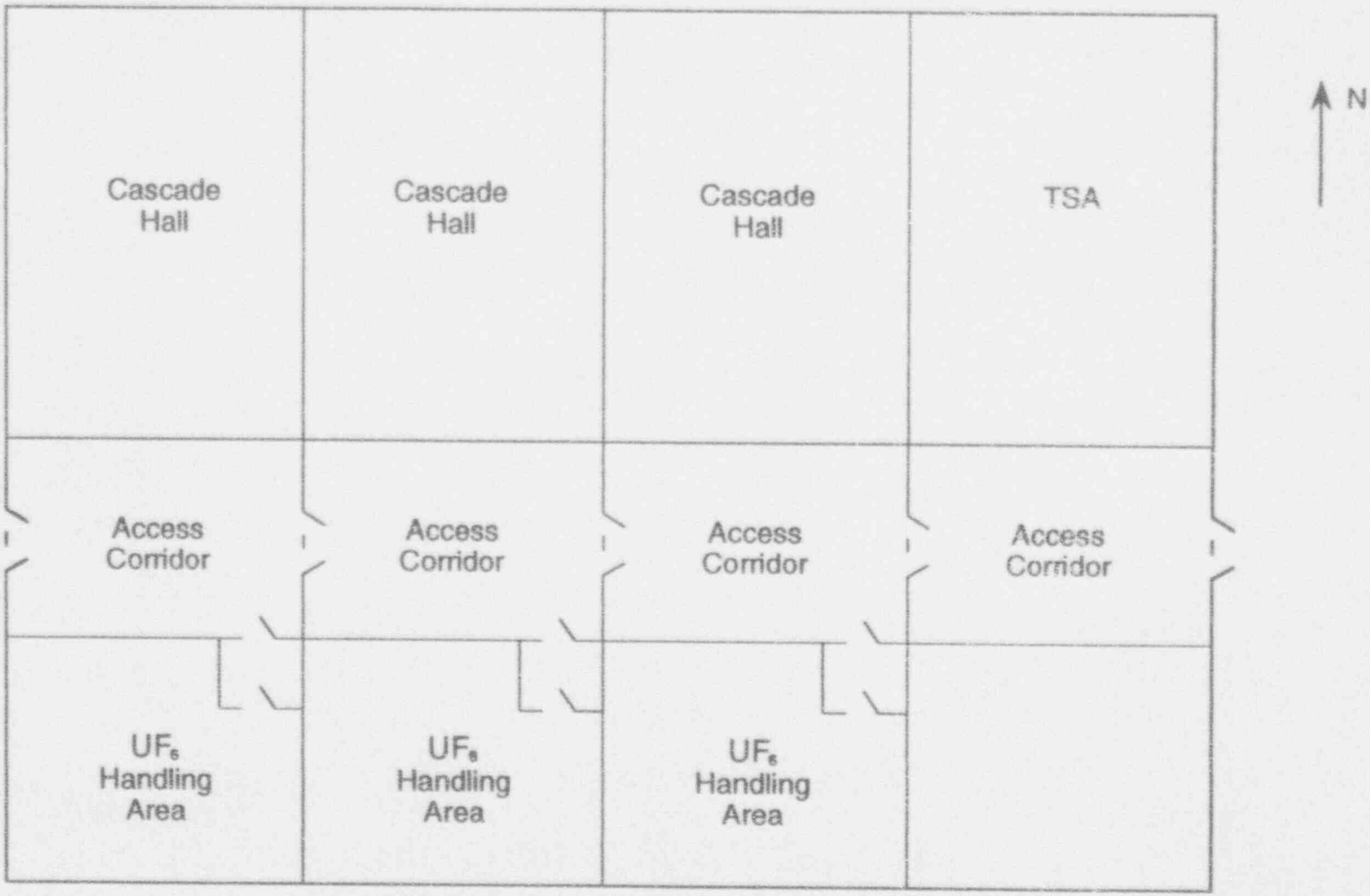

Figure 4.3 Separations Building leakpath flow schematic

access corridor and the referenced resistance coefficients, the NRC staff estimated a leakage flow from a UF_6 Handling Area of 0.014 m^3/s (30 cfm). As an additional conservatism, the analysis did not consider the reduction in contaminant concentration which would occur in the mixing of the UF_6 Handling Area leakage flow and the corridor leakage flow. In light of these considerations, Gaussian modeling is used for atmospheric dispersion modeling. The following paragraphs describe the dispersion modeling for the continuous and instantaneous release modes.

Dispersion Modeling for Continuous Releases

Regulatory Guide 1.145 (NRC, 1982b) provides guidance for applying Gaussian plume modeling of continuous releases and for selecting meteorological conditions representing accidents. The analysis followed this guidance in the screening procedure. It considered plume rise, building wake effects, and relative frequency of occurrence of wind speed, stability class, and wind direction. It calculated the frequency distributions of concentration per unit source (χ/Q) for each of sixteen direction categories and for all directions considered together, for both elevated and ground-level releases. For elevated releases, the calculations were repeated for a set of distances to identify the maximally exposed individual. For ground-level releases, the maximally exposed individual is located at the edge of the controlled area as χ/Q decreases with distance. The selected χ/Q values were used to identify release conditions which exceeded the NUREG-1391 criteria.

For elevated releases, estimated χ/Q values at a set of distances are summarized in Table 4.1. The results indicate that the maximally exposed individual is located in the northern sector at a distance of 400 meters and that the 95-percent-overall and the largest 99.5-percent-sector χ/Q values are equal to 1.7×10^{-5} s/m^3. The χ/Q value established by this analysis is used in conjunction with release durations less than or equal to two hours in order to calculate UF_6 release rates which would produce uranium doses and HF concentrations equal to the NUREG-1391 guidance adopted as the screening criteria. For continuous releases into the building space which exit through the stack, the off-site receptor is controlling, and the uranium dose criterion identifies 346 gm/s as the allowable UF_6 release rate. For the same release scenario, applying the HF concentration criterion would identify 3200 gm/s as the allowable UF_6 release rate. Consideration of release duration is an integral part of the analysis because the NUREG-1391 HF concentration criterion is specified as a function of exposure time.

Dispersion modeling estimates of χ/Q for continuous ground-level releases for the two potential maximally exposed individuals are summarized in Table 4.2. In this case the 95-percent-overall and the maximum 99.5-percent-sector χ/Q values are again approximately equal. χ/Q values adopted for the screening analysis are 6.9×10^{-3} and 1.5×10^{-3} s/m^3 for the fence line and 400-meter receptor locations. The 400-meter location is selected as a conservative representation of offsite conditions in order to maintain a set of receptor

Table 4.1 χ/Q Estimates for elevated continuous releases

Direction	χ/Q (s/m^3) Distance to Receptor		
	200 m	400 m	600 m
95-percent Overall			
	$8.4X10^{-6}$	$1.7x10^{-5}$	$1.3x10^{-5}$
99.5-percent Sector			
S	$8.1x10^{-6}$	$1.7x10^{-5}$	$1.2x10^{-5}$
SSW	$1.8x10^{-6}$	$1.1x10^{-5}$	$9.5x10^{-6}$
SW	$1.8x10^{-6}$	$1.1x10^{-5}$	$1.0x10^{-5}$
WSW	$1.7x10^{-6}$	$1.1x10^{-5}$	$9.7x10^{-6}$
W	$6.0x10^{-6}$	$1.4x10^{-5}$	$1.1x10^{-5}$
WNW	$1.8x10^{-6}$	$1.3x10^{-5}$	$1.1x10^{-5}$
NW	$7.2x10^{-6}$	$1.5x10^{-5}$	$1.1x10^{-5}$
NNW	$5.5x10^{-6}$	$1.4x10^{-5}$	$1.1x10^{-5}$
N	$9.2x10^{-6}$	$1.7x10^{-5}$	$1.4x10^{-5}$
NNE	$5.3x10^{-6}$	$1.3x10^{-5}$	$1.1x10^{-5}$
NE	$7.1x10^{-6}$	$1.4x10^{-5}$	$1.1x10^{-5}$
ENE	$7.0x10^{-6}$	$1.5x10^{-5}$	$1.1x10^{-5}$
E	$6.8x10^{-6}$	$1.4x10^{-5}$	$1.1x10^{-5}$
ESE	$1.7x10^{-6}$	$1.1x10^{-5}$	$9.6x10^{-6}$
SE	$1.8x10^{-6}$	$1.1x10^{-5}$	$9.9x10^{-6}$
SSE	$1.8x10^{-6}$	$1.1x10^{-5}$	$9.9x10^{-6}$

Table 4.2 χ/Q Estimates for continuous ground level releases

Direction	χ/Q (s/m^3) Distance to Receptor	
	165 m	400 m
95-percent Overall		
	$6.9X10^{-3}$	$1.5x10^{-3}$
99.5-percent Sector		
S	$1.9x10^{-3}$	$4.8x10^{-4}$
SSW	$1.3x10^{-3}$	$2.8x10^{-4}$
SW	$1.2x10^{-3}$	$2.7X10^{-4}$
WSW	$1.2x10^{-3}$	$2.7x10^{-4}$
W	$2.3x10^{-3}$	$5.7x10^{-4}$
WNW	$6.5x10^{-3}$	$1.5x10^{-3}$
NW	$7.4x10^{-3}$	$1.6x10^{-3}$
NNW	$7.4x10^{-3}$	$1.6x10^{-3}$
N	$7.0x10^{-3}$	$1.6x10^{-3}$
NNE	$4.0x10^{-3}$	$8.8x10^{-4}$
NE	$3.4x10^{-3}$	$7.5x10^{-4}$
ENE	$7.4x10^{-3}$	$1.6x10^{-3}$
E	$7.8x10^{-3}$	$1.7x10^{-3}$
ESE	$4.1x10^{-3}$	$9.1x10^{-4}$
SE	$1.2x10^{-3}$	$2.8x10^{-4}$
SSE	$5.1x10^{-3}$	$1.2x10^{-4}$

locations consistent with those identified in the elevated release analysis. UF_6 release scenarios involving continuous ground-level releases are leakpath releases resulting from continuous or instantaneous releases into the building space or continuous releases outside the building. In each case, the controlling receptor is located at the controlled area fence. For continuous releases into the building space, the uranium intake criterion is controlling, and the maximum allowable UF_6 release rate is 204 gm/s. For instantaneous releases into the building space followed by leakpath continuous release, the uranium intake criterion is limiting, and the allowable quantity is 1470 kilograms of UF_6. For continuous ground-level releases outside of the building, the uranium intake criterion is limiting, and the allowable release rate of UF_6 is 0.85 gm/s.

Dispersion Modeling for Instantaneous (Puff) Releases

Evaluation of uranium intakes and HF concentrations for comparison with the time dependent NUREG-1391 criteria requires estimation of release and exposure time intervals. In order to analyze all potentially important release scenarios and to explicitly evaluate potential time dependent impacts, the NRC staff evaluated instantaneous release scenarios. To maintain consistency with the continuous release analysis, the NRC staff adopted the general approach of Regulatory Guide 1.145 for use with a Gaussian puff model. The joint frequency of meteorological conditions is used to calculate χ/Q values, and the critical location is identified for both elevated and ground-level releases. Uranium intakes and HF concentrations which would not be exceeded 95 percent of the time are calculated for a unit release of UF_6. Because doses and concentrations are linear functions of the UF_6 release quantity for instantaneous releases, the analysis is used to identify release quantities which would exceed the criteria.

For uranium intakes from elevated releases, the critical receptor is located at a distance of 400 meters north of the stacks, and a release of 1,785 kilograms of UF_6 would produce an intake equal to the NUREG-1391 limit (10 milligrams). For HF concentrations from elevated releases, the critical location is 200 meters north of the stacks, and a release of 1,480 kilograms of UF_6 would exceed the NUREG-1391 criterion.

For uranium intakes from ground-level releases, the critical receptor is located at the fence line (165 meters), and release of 4.3 kilograms of UF_6 would produce a dose exceeding the NUREG-1391 criterion. For HF concentrations from ground-level releases, the critical receptor is located at the fence line, and a release of 2.8 kilograms of UF_6 would produce a concentration in excess of the NUREG-1391 criterion.

4.2.3.2 Results

Scenarios developed in simple form for first-stage analysis and in more complex form for second-stage analysis were used in conjunction with the dispersion analysis to identify structures, systems, and components important to safety. To provide an exhaustive review of hazards for the entire site, the screening procedure was implemented for each site area and building on an overall unit basis and on a system-by-system basis. The initial step in the analysis serves to review the function of the site areas and buildings, and to identify potential common-cause failures; the system-by-system review examines the function of smaller elements at a more detailed level. In the Separations Building review, the system-by-system review takes advantage of the similarity of Plant Units 1, 2, and 3 in the process from the feed system through enrichment to the take-off systems. The UF_6 blending, sampling, and storage systems are considered as separate plant elements. Hazards related to materials other than UF_6 are considered at each step of the procedure.

Separations Building Review

The Separations Building houses the major components of the enrichment system, including feed, sampling, and blending autoclaves processing UF_6 in the liquid state. The hazard audit identified inventories of liquid UF_6 in the UF_6 Handling and Blending Areas, inventories of solid UF_6 in the Cylinder Handling, UF_6 Handling, and the Blending Areas; and the flow of gaseous UF_6 in the UF_6 Handling Area and the Cascade Halls.

Systems Handling Liquid UF_6

Collapse of the building in an earthquake could damage the autoclaves and cylinders containing liquid UF_6 and thereby cause an instantaneous release from the rapid depressurization of the mixture of liquid and gaseous UF_6. Each plant unit has more than one feed autoclave and two blending autoclaves which may contain liquid UF_6 at a given time. However, structural analysis described in Section 4.4 demonstrates that the Separations Building does not collapse under design basis earthquake conditions. Thus, a release does not occur, and analysis of this scenario does not identify the structure as important to safety. Similarly, if an autoclave containing liquid UF_6 were to overturn or slide during an earthquake, loss of containment could occur. Analysis presented in Section 4.8 demonstrates the autoclaves do not slide or over-turn when subjected to design basis earthquake forces. Thus, analysis of this scenario does not identify a component important to safety. Similar considerations apply for scenarios involving the design basis tornado (DBT) and DBT-missiles as the initiating event. Analysis presented in Section 4.3 indicates that the Separations Building provides adequate protection under DBT conditions.

UF_6 is held in the liquid state in feed and blending autoclaves after completion of the cold and hot purification cycles. During this standby period, because valves in the autoclave exit line are closed, UF_6 heating could cause over-pressurization of the cylinder and autoclave. Similarly, the contents of the sampling autoclave are held at elevated temperature and pressure, and not vented. Ruptures of these cylinders and autoclaves, with a sudden loss of pressure, would lead to large releases. If the final state is an equilibrium mixture of vapor and solid at the atmospheric pressure sublimation point, the estimated amounts of UF_6 vapor produced in the ruptures of single feed or product cylinders are 9,425 kilograms and 1,510 kilograms of UF_6, respectively. In order to provide conservative estimates of potential UF_6 releases, the initial temperatures and pressures used in the calculations were the upper level set points of the heater protection circuits. These quantities are in excess of the 1,470 kilogram screening criterion. Therefore, the heater components and control circuits, and the autoclave temperature and pressure sensors and associated controls (TE-122, TE-127, PT-115 and PT-118) of the feed, blending, and sampling autoclaves are systems and components important to safety.

Systems Handling Solid UF_6

Cylinders containing solid UF_6 are stored in the Cylinder Handling Area on a temporary basis before transfer to the UF_6 Handling Area. An accident involving a fuel spill and subsequent fire could cause over-pressurization and rupture of a UF_6 cylinder. Rapid de-pressurization of the cylinder during failure would produce an instantaneous release greater than the 1,470 kilogram screening criterion. Thus, transporters carrying significant quantities of fuel or the UF_6 cylinders could be considered components important to safety. However, occurrence of this scenario is prevented by limitation of transporter fuel inventory as proposed by the applicant (LES, 1993e).

Cylinders containing solid UF_6 may be ruptured if dropped in handling or crushed in vehicle collisions. Such events could expose the solid UF_6 to the building air, and result in the gradual sublimation of the UF_6 and subsequent production of HF. A conservative estimate of the sublimation rate was developed by using mass and energy balances formulated for a cylindrical mass of UF_6 exposed to the atmosphere. Mass and heat transfer coefficients were estimated by using standard correlations of experimental data (Bird, Stewart, and Lightfoot, 1960). The UF_6 release rate predicted in this fashion is 0.016 gm/s and is too low to threaten public health and safety in a reasonable release duration.

The feed, product take-off, and blending systems use desublimers in which UF_6 is normally present in solid and gaseous states. During the transition from the solid to gaseous state, UF_6 is heated with Freon refrigerant R-11 at 122 °F. Desublimer inlet and outlet valves are closed during this heating period. If the desublimer is over-filled, the solid UF_6 can expand and rupture the desublimer pipe. The sudden loss of pressure would result in evolution of approximately 10 kilograms of UF_6 vapor, with the balance of the pipe inventory remaining in the solid state. Failure of the desublimer tube would also damage the Freon coil and terminate return flow to the Freon supply system and the heating of the UF_6. The quantity of UF_6 generated because of heat transfer through the desublimer cabinet would be minimal. Accordingly, the desublimers are not important to safety on the basis of this potential failure mode.

Solid UF_6 in the feed, product take-off, and blending system desublimers is heated with recirculated Freon to help transfer UF_6 to the appropriate cylinders. The Hot Refrigerant Supply System in each plant unit indirectly heats the Freon with water, and a controller limits the water temperature to 165 °F. If the Freon temperature sensor fails and the water temperature control system functions properly, Freon could be supplied to the desublimers at 165 °F. Although the failure of the desublimer tube is not an important safety condition, overheating and overfilling which cause a failure with the UF_6 in the liquid state is a potentially more hazardous situation. A rupture produced by these conditions is equivalent in its effect to the rupture of a single desublimer tube containing 2,420 kilograms of UF_6. Rapid de-pressurization of a desublimer tube from the liquid state at 165°F produces approximately 1,100 kilograms of gaseous UF_6. Because this amount is below the criterion amount, the analysis does not identify components in single-tube desublimers important to safety. For the feed desublimers, which contain four tubes, the analysis indicates that operator attention to the change of weight of feed cylinders is required to maintain a safe condition. Because approximately 100 kilograms of UF_6 is transferred to the desublimer in purification of a single cylinder, the contents of multiple cylinders would have to be transferred to the desublimer to reach a potentially dangerous condition. Because credit is given for attention by multiple operators over an extended period of time, the NRC staff analysis of this scenario does not identify a system as important to safety.

Carbon and alumina traps are used in the desublimer vent systems and in the Gaseous Effluent Vent System (GEVS) to adsorb UF_6 and HF. These systems have uranium capacities lower than the 1470-kilogram limit and, thus, are not important to safety.

Systems Handling Gaseous UF_6

If the Separations Building structure and autoclaves in each plant unit remain functional in all design basis events but all three plant unit feed headers fail, a maximum continuous release of approximately 150 gm/s is possible. The dispersion and scenario analysis concluded that a continuous release of at least 204 gm/s is required to exceed the NUREG-1391 criterion. Thus, the CEC pipework is not identified as a system important to safety. In addition, each of the feed headers has a pressure sensor (PT-113) and associated circuitry designed to shut valve HV-134 located inside the autoclave. Because closing these valves prevents the release, the sensors, control circuitry, and valves provide additional assurance that the hazardous condition will not occur.

The hazard audit identified the gaseous UF_6 inventory of the cascades as a total of 420 kilograms, a quantity less than the 1,470-kilogram instantaneous release criterion. Thus, the centrifuges and associated piping and control systems are not identified as systems or components important to safety on the basis of inventory. The flow system review identified the maximum inlet flow to any one cascade as approximately 3.5 gm/s. A release interval greater than 200 hours is required at this release rate to produce a dose in excess of the screening guidelines. Because operator response to a number of indications of abnormal flow is credited, individual systems and components in the cascades are not identified as important to safety on the basis of flow.

The flow system review identified the product and tails take-off headers as handling a total UF_6 flow of 150 gm/s. As in the case of the feed headers, simultaneous failure of these headers does not produce a release in excess of the review criterion. Failure of multiple cascade components other than the exit headers also produces release flow rates too low to require safety protection systems.

The flow system review determined that the maximum UF_6 flow from the blending autoclaves is approximately 75 gm/s. This flow rate is lower than the 204 gm/s level required to exceed the NUREG-1391 criterion. In addition, the dispersion analysis indicated that reaching the criterion would require a release period of greater than five hours. At this release rate, the HF monitors in the UF_6 Handling Area and the alpha activity monitors in the ventilation system would be activated. Because of the low level of the release and credit given for operator response to multiple alarms, a continuous release of this magnitude and duration does not identify any components important to safety.

Storage Area and Cylinder Receipt and Dispatch Building Reviews

Cylinders containing solid UF_6 are stored in the Product, Feed, and Tails Storage Areas and are handled in the Cylinder Receipt and Dispatch Building. Accidents which could take place in these areas include the collision, dropping, and fire scenarios like those of the Separations Building Cylinder Handling Area. Analysis of these events would identify vehicles carrying fuel which sustains a substantial fire or the cylinders themselves as components important to safety. However, occurrence of this scenario is prevented by limitation of transporter fuel inventory as proposed by the applicant (LES, 1993e).

4.3 Wind and Tornado Design

The ANPR for enrichment facilities requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena including the DBT and DBT-generated missiles without loss of any safety functions. The NRC staff evaluated the response of the Separations Building to such effects by reviewing the applicant's analyses presented in the CEC SAR (LES, 1993a) and supporting documents (LES, 1991), and the design basis criteria for compliance to the applicable codes and standards. The NRC staff examined the analysis reports and their calculations for correctness, and determined whether the structure can retain its integrity against the effects of natural phenomena and thereby protect public health and safety. This section presents a summary description of the structures, identifies design basis criteria, evaluates acceptability of the analytical methods, and compares the resulting forces and stresses to the appropriate allowable limits.

The Separations Building is divided into three independent plant units each of which is comprised of a UF_6 Handling Area, Auxiliary Area, and Cascade Halls. A Technical Services Area supports activities in all three plant units. The Separations Building structure is System Class II, and its Quality Assurance level is 2. The members and the components of the Separations Building are cylindrical steel stacks, rectangular concrete columns, solid concrete walls, precast/prestressed concrete beams, and double-tee roof and floor members.

4.3.1 Design Criteria

CEC SAR Section 4.2.2 details the design criteria. This SAR included a site-specific study to determine the tornado design parameters. The requirements of ANSI A58.1, Section 6, were adopted in SAR Section 4.2.1 and used for developing tornado pressure loadings. The NRC staff accepts the design criteria. The tornado-generated missiles postulated in the analysis are a 15-pound, 2x4-inch wood plank traveling 100 mph and a 75 pound, 3-inch steel pipe traveling 50 mph.

The load combinations identified in SAR Sections 4.2.8.4.1. and 4.2.8.4.2. for the Class I concrete-and-steel structures were compared to the requirements of AISC, ACI-318, ACI 349-85, and ANSI/ANS-57.9-84. The combinations agree with the codes and standards, and the NRC staff finds them to be acceptable. The results of the comparison of the NRC staff's and applicant's load combinations are shown in Tables 4.3 and 4.4. At the design review stage, all the combinations need not be used. The factor-of-safety values for overturning and sliding of foundations were given in Section 4.2.8.4.3 of the SAR. The NRC staff's evaluations are shown in Table 4.5.

4.3.2 Design Evaluation

The applicant conducted analyses of the required grout on the roof members and the concrete wall thickness to satisfy the SAR requirement for preventing scabbing or penetration by tornado missiles, as reported in DC-SE-0001-SD, REV.1 (LES, 1991). The NRC staff checked these calculations and found the results satisfactory, as indicated in Table 4.6; the staff also checked the tornado forces and missile loads on the exterior walls and found them to be acceptable, as shown on Table 4.7. The sizing of the roof members, interior and exterior columns, and shear walls of the Separations Building is given in report DC-SE-0003-SD, REV.0 (LES, 1991). The preliminary design of roof members, beams, and interior columns is performed for the combination of live load and dead load. Exterior columns and shear walls were designed to the DBT. For shear walls, the design-basis earthquake (DBE) condition dominates the DBT condition. The NRC staff's and applicant's estimated shearwall forces are compared in Table 4.8. The NRC staff concludes that the Separations Building design is structurally acceptable for the natural phenomenon loads.

The design calculations for the stacks are included in reports DC-SE-0007-SD, REV. 0 and DC-SE-0001-SD, REV. 0 (LES, 1991). The results of critical stress calculations for stacks are tabulated in Table 4.9. The dead load of the stack was increased 20 percent in order to account for ladder and plant forms. The NRC staff concurs with the SAR conclusion that the DBT, not the DBE condition, controls the design. The qualification of the section, anchor bolts, base plate, and foundation footing for the load combination of S=D+L+WT was performed for 1/4-inch-thick wall stacks. The NRC staff determined that the 1/4-inch wall thickness originally proposed for the stacks was insufficient to meet DBT requirements. The governing load combination is 1.6S=D+L+WT+missile. The minimum wall thickness was

Table 4.3 Load combination for class I concrete structure

Combination	LES (Sec. 4.2.8.4.1. of SAR)	NRC	NRC Ref./Comments
a	U=1.4D+1.7L+1.7(LR or S or R)+1.7H+1.4F		Acceptable
b	U=1.4D+1.7L+1.7H+1.4F+1.7W	U=1.4D+1.7L+1.7H+1.4F +1.7W+1.7Ro	Acceptable. Ref. ACI 349-85 Sec. 9.2.1-(3). LES is missing Ro.
c	U=1.4D+1.7L+1.7H+1.4F+1.87E	U=1.4D+1.7L+1.7H+1.4F+1.7E+1.7Ro	Acceptable. Ref. ACI 349-85 Sec. 9.2.1-(2). LES is missing Ro.
d	U=.75(1.4D+1.7L+1.7H+1.4F+1.4T+1.87E)		Acceptable
e	U=.75(1.4D+1.7L+1.7H+1.4F+1.4T+1.87E)	U=.75(1.4D+1.7L+1.7Ro+1.4T+1.9E)	Acceptable. Ref. SRP 3.8.4-II.3.b.(5)
f	U=.75(1.4D+1.7L+1.7H+1.4F+1.4T+1.7W)	U=.75(1.4D+1.7L+1.7H+1.7T+1.7W)	Acceptable. Ref. ANSI/ANS-57.9-84. Sec. 6.17.3.1-(c). Code has 1.7T and no 1.4F
g	U=.9D+1.3W	U=1.2D+1.7W	Acceptable.
h	U=.9D+1.43E	U=1.2D+1.9E	Acceptable.
extreme environmental conditions:			
i	U=D+L+T+E'	U=D+L+Ro+T+E'	Acceptable. Ref. SRP 3.8.4-II.3.(i).b.(8). Ro is missing.
h	U=D+L+T+Wt	U=D+L+Ro+T+Wt	Acceptable. Ref. SRP 3.8.4-II.3.(i).(7), Ro is missing.
j	U=D+L+T+DBFL	U=D+L+T+F+Ro+1.25Pa	Acceptable. Ref. ACI 349-85, Sec. 9.2.1-(6). To, Ro, Pa are missing.

LOAD NOMENCLATURE:

D - Dead load.
L - Live load.
W, Wt - Wind & tornado loads.
E, E' - Design basis earthquake.

T - Thermal.
H - Lateral soil pressure.
F - Lateral and vertical pressure of liquids (DBFL).
Ro - Piping and equipment reactions.
Pa - Forces generated by a postulated pipe break.

Table 4.4 Load combination for system class I steel structure

	LES (Sec. 4.2.8.4.2 of SAR)	NRC	NRC Ref./Comments
a	1.0S=D+L+(LR or S or R)		
b	1.0S=D+L+W	1.0S=D+L+W	Acceptable. SRP 3.8.4-II-c.(a).(3)
c	1.0S=D+L+E	1.0S=D+L+E	Acceptable. SRP 3.8.4-II-c.(a).(2)
d	1.5S=D+L+T		Acceptable
e	1.5S=D+L+T+W	1.5S=D+L+T+H+W	Accept. ANSI/ANS.57.9-84, Sec. 6.17.3.2.1.(d) Code has H in combination but H = 0.
f	1.5S=D+L+T+E	1.6S=D+L+T+H+E	Accept. ANSI/ANS.57.9-84, Sec. 6.17.3.2.1.(e) Code is 1.6S and has H in combination but H = 0.
extreme environmental conditions:			
g	1.6S*=D+L+T+E'	1.6S=D+L+T+H+E'	Accept. ANSI/ANS.57.9-84, Sec. 6.17.3.2.1.(e) Code has H in combination but H = 0.
h	1.6S*=D+L+T+DBFL	1.5S=D+L+T+H+Wt	Not accept. ANSI/ANS.57.9-84, Sec. 6.17.3.2.1.(d) Code has H in combination but H and DBFL = 0.
i	1.6S*=D+L+T+DBFL	-	No ref.

*The allow. stresses cannot exceed .7Fu and .7Fu Z/Zp (Zp is plastic section modulus).

Table 4.5 Load combination safety factors for foundations

Combination	LES (Sec. 4.2.8.4.3 of SAR)		NRC (Ref. ANSI/ANS-57.9-84, Sec. 6.17.4)		NRC Comments
	Overturning	Sliding	Overturning	Sliding	
D+H	-	-	1.5	1.5	No value by LES but encompassed by D+H+E combination.
D+H+E	1.5	2.0	1.1	1.1	Acceptable
D+H+W	1.5	2.0	1.1	1.1	Acceptable
D+H+E'	1.5	2.0	-	-	Acceptable, but no ref. was found.
D+H+Wt	1.5	2.0	-	-	Acceptable, but no ref. was found.

LOAD NOMENCLATURE:

D - Dead load.
L - Live load.
W, Wt - Wind & tornado loads.
E, E' - Design basis earthquake.
T - Thermal.
H - Lateral soil pressure.
F - Lateral and vertical pressure of liquids (DBFL).
Ro - Piping and equipment reactions.
Pa - Forces generated by a postulated pipe break.

Table 4.6 Required exterior concrete thickness to resist tornado missiles: Ref. Calc. DC-SE-0001-SD, Rev. 1

Missiles	2X4 Timber		3" Steel Pipe		NRC Comments
	LES	NRC	LES	NRC	
Concrete wall	100 mph	100 mph	75 mph	75 mph	Acceptable
Penetration	1.28"	1.28"	1.81"	1.81"	Acceptable
Scabbing	6.09"	6.09"	6.04"	6.04"	Acceptable
Required thick.	8.28"	8.29"	7.24"	7.24"	Acceptable
Thick. used	-	-	8.00"	8.00"	Acceptable (timber is deformable)
Concrete roof	70 mph	70 mph	35 mph	35 mph	Acceptable
Penetration	.98"	.98"	1.39"	1.39"	Acceptable
Scabbing	5.87"	5.87"	5.46"	5.46"	Acceptable
Required thick.	7.04"	7.04"	6.55"	6.55"	Acceptable
Thick. used	-	-	6.50"	6.50"	Acceptable (timber is deformable)

Table 4.7 Tornado forces on exterior panels: Ref. Calc. DC-SE-0001-SD, Rev. 01

Exterior Panel	36 ft Panel		47 ft Panel		NRC Comments
	LES	NRC	LES	NRC	
Outward wind pressure	63.6 psf	63.6 psf	63.2 psf	63.2 psf	Acceptable
Inward wind pressure	20.0 psf	20.0 psf	20.7 psf	20.7 psf	Acceptable
Missile crushing load (inward)	26.8 kips	26.8 kips	26.8 kips	26.8 kips	Acceptable
Moment at center (with missile)	59.2 ft-kips	59.2 ft-kips	81 ft-kips	81 ft-kips	Acceptable
Panel moment capacity at center	69.2 ft-kips	69.2 ft-kips	138.4 ft-kips	138.4 ft-kips	Acceptable
Moment at load (with missile)	105.8 ft-kips	105.8 ft-kips	145.9 ft-kips	145.9 ft-kips	Acceptable
Panel moment capacity at load	155.7 ft-kips	155.7 ft-kips	203.3 ft-kips	203.3 ft-kips	Acceptable
Max. shear stress.	51.6 psi	51.6 psi	59.2 psi	59.2 psi	Acceptable
Panel shear allow.	141 psi	141 psi	141 psi	141 psi	Acceptable
ratio=stress/allow	.37	.37	.42	.42	Acceptable

Table 4.8 Total DBT and DBE shear wall forces for separations building

Structure	Cascade Hall		UF6 Building		NRC Comments
	LES	NRC	LES	NRC	Acceptable
E-W Design wind pressure	51.0 psf	51.0 psf	45.3 psf	45.3 psf	Acceptable
Total E-W DBT load	343 kips	343 kips	163 kips	163 kips	Acceptable
N-S Design wind pressure	60.3 psf	60.3 psf	56.1 psf	56.1 psf	Acceptable
Total N-S DBT load	859 kips	859 kips	612 kips	612 kips	Acceptable
Total DBE load (in the direction of one major axis)	2160 kips	3645 kips	3191 kips	3191 kips	Acceptable

Table 4.9 Stress on stack: Ref. Calc. DC-SE-0007-SD, Rev. 0

	64" Stack		36" Stack		NRC Comments
	LES (t=1/4")	NRC (t=3/8)	LES (t=1/4")	NRC (t=3/8")	
Max. wind press.	55 psf	55 psf	55psf	55 psf	3/8" Acceptable
Max. wind for./ft	293 plf	293 plf	165 plf	165 plf	3/8" Acceptable
Wind shear @ roof level	21.2 kips	21.2 kips	12.0 kips	12.0 kips	3/8" Acceptable
Max. wind moment @ roof level	812 ft-kips	812 ft-kips	460 ft-kips	460 ft-kips	3/8" Acceptable
Missile load @ top 2X4 timber 3" pipe	 - -	 2.7 kips 6.8 kips	 - -	 1.51 kips 3.78 kips	 3/8" Acceptable 3/8" Acceptable
Worst bending stress (D+L+Wt+Missile)	-	13.35 ksi	18.6 ksi	24.1 ksi	3/8" Acceptable
Allow. bending stress Fb=1.6*.66*S FB=1.5*.66*S	 34.98 ksi 	 29.29 ksi -	 34.98 ksi 	 32.79 ksi	3/8" Acceptable
ratio= stress/allowable	-	0.45	0.53	0.7	3/8" Acceptable

subsequently increased by the applicant to 3/8 inches to prevent local perforation of a stack steel plate because of tornado missile impact. The applicant's calculation to justify a 3/8-inch thickness is shown in the report DC-SE-0001-SD, REV.1. The NRC staff accepted the analysis with the revised 3/8-inch wall thickness. The applicant's tornado-generated missile impact analysis which evaluated only cross-sectional bending considered a 2x4-inch timber striking the top of the 36-inch-diameter stack. The NRC staff performed calculations to determine the severity of the 3-inch pipe impact. As reported in Table 4.9, the additional moment caused by the 3-inch diameter steel pipe is larger than the moment caused by the 2x4-inch wood board. As a result, the NRC staff checked the actual bending stress in the two stacks for the DBT and 3-inch pipe missile impact and found the resulting stresses to be within the allowable stress level at the normal metal temperature. On the basis of a review of the submitted documentation, the NRC staff accepts this stress analysis.

4.4 Water Level (Flood) Design

Catastrophic floods cannot cause a release of UF_6 because the site is not located in or near a major floodplain or below a large body of surface water, as described in Section 2.4.3. An evaluation was made to determine if releases of UF_6 could occur as a result of local intense precipitation.

The applicant adopted the Standard Project Flood (SPF) as the CEC design basis flood in accord with ANPR guidance (NRC, 1988a). The applicant conservatively assumed the SPF to be equal to 68 percent of the Probable Maximum Flood (PMF), a percentage exceeding the U.S. Army Corps of Engineers (COE) guidance of 40 to 60 percent of the PMF (Dept. of the Army, 1964). From information in National Oceanic and Atmospheric Administration's (NOAA's) Hydrometeorological Report (NOAA, 1978), the applicant estimated that the resulting Standard Project Storm (SPS) produces an accumulation of 2.5 inches of water in the plant yard (LES, 1993a). Safety Class I structures are constructed at least 6 inches above the yard grade elevation. Because of site surface hydrologic conditions, site drainage, and placement of facilities, the PMF will have no effect on UF_6 stored or used at the CEC. The NRC staff has reviewed the applicant analysis of the effect of the design basis flood and concludes that the flood would not pose a threat to operation of the facility or to public health and safety.

4.5 Seismic Design

This section summarizes the description of the structures and identifies the seismic design-basis criteria, evaluates the analysis methods, and compares the resulting forces and stresses to the appropriate allowable limits.

The ANPR (NRC, 1988a) requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. The NRC staff's evaluation approach is to review the CEC SAR (LES, 1993a) and design-basis criteria for compliance with the applicable codes and standards, check the analysis reports and calculations for correctness, and determine whether the structure can retain its integrity against the effects of earthquakes and thereby protect public health and safety.

4.5.1 Earthquake Design Criteria and Acceptability

The applicant used the standard building code (SBCCI, 1988) and DBE analytical methods to evaluate the CEC design criteria. The design criteria are addressed in SAR Section 4.2.5.

The standard building code earthquake site-specific parameters for buildings are given in SAR Section 4.2.5.1. This approach applies a total lateral force in the direction of the main axes of the structure and applies forces with 5-percent planar eccentricity per code requirements. A load factor of 1.5 is applied to the earthquake forces in the ultimate design method approach used. The results were compared to the member capacity.

The applicant used the equivalent static load method to perform the design basis earthquake analysis. The acceleration values used in the analysis were determined from the response spectra curves by using the calculated fundamental frequencies of the structures. Damping values used are 5 percent for the stacks and 10 percent for the concrete Separations Building. The appropriate acceleration values were applied, and the resultant member forces were compared to the allowable member limits. At the first mode, total weight participation was assumed. The NRC staff compared the damping values presented in CEC SAR Table 4.2-1 with those in the Regulatory Guide 1.61 guidance. It should be noted that the NRC staff did not concur with the high damping values selected by the applicant. In the NRC staff review, lower damping values were used. The findings are summarized in Table 4.10.

4.5.2 Analytical Results

The Separations Building and the gaseous effluent vent stacks are designed to the standard building code and the DBE requirements, as stated in report DC-SE-0003-SD, REV 0 (LES, 1991). The NRC staff checked the analytical results for the Separations Building against the standard building code for one major axis. The results are acceptable, as shown in Table 4.11.

The CEC SAR does not justify representing the structures by a simple model, and the required factor of 1.5 was not applied to the peak acceleration value in order to determine the total lateral shear force for the equivalent static load method. Further, the SAR does not include any combination of orthogonal seismic forces to determine the maximum resultant shear force on shear walls. The NRC staff performed calculations to determine the maximum resultant forces by combining two lateral orthogonal seismic forces by the Square Root of the Sum of the Squares method. Peak acceleration values at 5-percent damping were used in the NRC staff's analysis. The NRC staff's calculation results were compared with code requirements and are shown to be within the allowable limits. Thus, the NRC staff accepts the approach of using a simplified model for this design review. The results of the NRC staff's review are satisfactory, as shown in Table 4.12.

Table 4.10 Damping values for seismic design (percent of critical damping)

Structures or Components	Damping values (%)		NRC Comments
	LES (Table 4.2-1 of SAR)	NRC (Ref. US R.G. 1.61)	
Equipment	3	3, for SSE	Acceptable
Piping	2, freq.> 20 Hz	2, for SSE	Acceptable
	5, freq.< 10 Hz	-	No ref. publication was found to justify 5% damping < 10Hz.
Steel frame structure	7 -	7, for SSE & bolted 4, for SSE & welded	Acceptable No value by LES
Reinforced concrete struc.	10	7, for SSE	NRC used 5% in calc.'s
Prestressed concrete struc.	7	5, for SSE	NRC used 5% in calc.'s
Masonry structures (concrete)	7	12	Acceptable. DOE UCRL-15910, Table 4-5
Dual systems	The value for the system with the higher damping value may be used	-	The value for the system with the lower damping value should be used

Table 4.11 SBC earthquake analysis results: Ref. Calc. DC-SE-0003-SD, Rev. 0

Structure	Cascade Hall		UF6 Building		NRC Comments
	LES	NRC	LES	NRC	
Lateral force coef. (ZIKCZ)	0.032	0.032	0.032	0.032	Acceptable
Structure Weight	27,000 kips	27,000 kips	23637 kips	23637 kips	Acceptable
Earthquake force	864 kips	864 kips	756 kips	756 kips	Acceptable
Critical shear wall	east wall	east wall	east wall	east wall	Acceptable
Max. shear in wall	323 kips	323 kips	236 kips	236 kips	Acceptable
Shear allowable	2178 kips	2178 kips	1523 kips	1523 kips	Acceptable
ratio= force/allowable	0.16	0.16	0.15	0.15	Acceptable

Table 4.12 DBE analysis results: Ref. Calc. DC-SE-0003-SD, Rev. 0

Structure	Caskade Hall		UF6 Building		NRC Comments
	LES	NRC	LES	NRC	
Building natural frequency	3.5 Hz	3.5 Hz	-	-	Acceptable
Accel. from response spectrum	.08 g for 10% demp.	.09 g for 5% damp.	.08 g for 10% demp.	.09 g for 5% damp.	Acceptable. (peak accel. values are close)
Structure weight	27,000 kips	27,000 kips	23637 kips	23637 kips	Acceptable
Earthquake force	2160 kips	2430 kips	1891 kips	2127 kips	Acceptable
Critical shear wall	east wall	east wall	east wall	east wall	Acceptable
Max. shear in wall	620 kips	1479 kips	454 kips	1083 kips	Acceptable. (two orthogonal forces SRSS'ed)
Shear allowable	2178 kips	2178 kips	1523 kips	1523 kips	Acceptable
ratio= force/allowable	.28	.68	.30	.71	Acceptable

The applicant used a dynamic analysis approach for the 64-inch stacks as described in Report DC-SE-0007-SD (LES, 1991). The NRC staff questions the use of the amplification factor of 3 in the calculations; however, the NRC staff agrees that the DBE condition does not control the design for the stacks and considers the stack designs adequate for the natural phenomenon hazard of tornado. Analysis of the DBT is presented in Section 4.3. The results of the DBT review are presented in Table 4.9.

4.6 Fire Protection

Program Management

The applicant's submittal describes features of a facility, and of a program and equipment for fire protection to operate that facility in a firesafe manner. The responsibility for fire protection rests with the Technical Support Superintendent. He is assisted by the Industrial Safety Manager, whose direct responsibility is to ensure the safe operation of the facility in accordance with occupational safety and health regulations, including fire protection. The applicant has established a Facility Safety Review Committee (FSRC), whose responsibilities include fire safety. This committee performs an annual audit of the facility, reviews proposed changes, and offers professional advice on safety issues. The committee reports to the facility manager.

Fire Hazard Analysis

The applicant has performed a fire hazard analysis of the facility and states that the design and installation of its fire protection equipment are guided by the analysis. The analysis follows the methodology in the National Fire Protection Association's (NFPA) Fire Protection Handbook. It assumes anticipated inventories of combustible materials and their proximity to the Cascade Halls and UF_6 Handling Areas of the Separations Building. It also assumes the effectiveness of fire barriers, but no active mitigating measures, such as fire suppression actions. The applicant states that the analysis demonstrates that postulated fires in the most likely locations would not damage safety class equipment or cause UF_6 release.

Fire Protection Features

The production buildings of the facility are constructed of noncombustible or limited combustible materials. The principal process building, the Separations Building, has a structure, including roof, beams, columns, and floor members and wall panels, made of precast/prestressed concrete construction. Fire barriers rated at 1 hour of fire resistance compartmentalize the Cascade Halls into three groups. The electrical distribution areas, UF_6 Handling Areas, laboratories, workshops, and storage areas are likewise separated by 1-hour-rated fire barriers. Doors and other apertures, such as for conduits, cable raceways, and ventilation ducts, in fire-rated assemblies are also fire-rated at least equal to the assemblies of which they are parts. The building is stated to have been designed to comply with the requirements of the NFPA code, NFPA 101, Life Safety Code.

Each of the two auxiliary buildings in which radioactive or contaminated materials may be handled, the Centrifuge Assembly Building and Cylinder Receipt and Dispatch Building, has a structure consisting of a braced steel frame enclosed in insulated metal siding and noncombustible or limited combustible roof, walls, and floor. The Standby Diesel Generator Building is similarly constructed of noncombustible or limited combustible materials.

All of these buildings, as well as the Office and Security Buildings and the Fire Pump House, are protected by automatic, wet-pipe sprinkler systems, except for the Central Control Room in the Separations Building and certain other areas containing electrical switchgears and batteries, which are protected by automatic pre-action sprinkler systems. Water deflectors or enclosures are provided in certain areas presenting a potential hazard of criticality. The sprinkler systems are stated to have been designed and tested in accordance with NFPA 13, Standard for the Installation of Sprinkler Systems.

Yard Storage of UF_6

The risk of a fire in the yard storage of UF_6 cylinders is reduced by administrative control of combustible materials in the area, use of limited-capacity fuel tanks for the forklift trucks, and by sloping the yard 0.2 percent toward drain systems.

Fire Water System

The facility has a fire water system, comprised of two 125,000-gallon tanks, two 1,000-gallon-per-minute fire pumps--one electric and one diesel engine driven--and a looped fire main with an adequate number of fire hydrants. The system is capable of supplying any of the sprinkler or standpipe systems and has a built-in redundancy in the water storage and pumping equipment. In addition, portable fire extinguishers are located throughout the facility.

Fire Alarm System

A facility-wide fire alarm system includes a central alarm console located in the Central Control Room, which is staffed around-the-clock, and local alarm panels in several buildings. An alarm can originate from a fire detector, a discharging sprinkler head, or a manually operated pull-box.

Fire Protection Equipment Maintenance

A surveillance and maintenance program works to ensure that fire protection equipment remains available and operable when needed in an emergency. Where applicable, the inspection, testing, and maintenance of the equipment will comply with industry standards, such as the NFPA codes.

Pre-Fire Plan and Fire Brigade

The applicant has developed a Pre-Fire Plan for use by the facility fire brigade. The brigade members are trained and equipped to respond to fire emergencies and contain fire damage until offsite help from a neighboring fire department arrives. The training program includes semi-annual refresher lessons and drills.

Conclusion

In consideration of the construction features of the facility, the applicant's commitment to implement a fire protection program and measures to respond to fire emergencies, and the proposed license conditions, the NRC staff concludes that the operation of the facility will be adequately firesafe.

4.7 Mechanical Systems and Components

SER Section 4.2.1 and 4.2.2 classify structures, systems, and components preventing or mitigating accidents which could expose off-site individuals to elevated levels of uranium and HF, as important to safety. The NRC staff's analysis identifies the feed, sampling, and blending autoclave heater protection circuits as mechanical components which are important to safety. The NRC staff's and applicant's analyses conclude that the Separations Building provides protection from wind loadings, tornado loadings, tornado-driven missiles, snow loadings, and rainfall loadings, that is, externally applied loads. Therefore, the only natural phenomenon potentially affecting the mechanical systems is the DBE.

SAR Section 4.2.7 discusses the magnitude of loads associated with process- and equipment-derived loadings. These include special pieces of equipment weighing more than 1,000 pounds, piping, HVAC and electrical tray and conduit loads reacted by the building and foundation. Design calculation DC-SE-0005-SD (LES, 1993c) presents the analysis for the foundations for the three types of autoclaves.

SAR Section 4.2.9 describes load combinations for mechanical equipment, piping systems, and HVAC systems which are in the important-to-safety category. The applicant provided design calculation DC-NT-IIC2-AA (LES, 1993c) to examine the effects of the DBE on the autoclaves. Any pressure vessels associated with the CEC, including autoclaves, are designed according to the ASME Boiler and Pressure Vessel (B&PV) Code Section VIII.

SAR Section 4.2.10 describes the criteria for load combination for electrical and control systems which are designated as important to safety. Each of the three autoclave types has important to safety air temperature and pressure control loops, safety control panels, and safety control panel supports as shown in SAR Table 4.6.1. These electrically related important-to-safety systems are analyzed in the design calculation DC-SE-008-SD (LES, 1993f).

4.7.1 Determination of Autoclave Response to the Design Basis Earthquake

Design calculation DC-NT-IIC2-AA (LES, 1993c) analyzes the response of the autoclaves to the DBE. This section of the SER discusses and evaluates the merits of the design calculation. As a result of NRC requests for additional information on the liquid sampling autoclave, the applicant submitted design calculation 04539001 (LES, 1993d) and additional documentation (LES, 1993g) to show that the tilting mechanism is designed to perform its function during and after the DBE.

The ANPR requires that components must be designed to withstand a DBE corresponding to a mean return period of 500 years. Applicant analysis accepted by the NRC staff identified the peak ground acceleration as 0.046g. The ANPR requires that components be designed by using a suitable dynamic analysis except where it can be demonstrated that an equivalent static load method provides adequate conservatism. The SAR did not contain such a demonstration. It did, however, use equivalent static load methodology.

Design calculation DC-SE-005-SD (LES, 1993c) makes a determination that the natural frequencies of product blending and feed autoclaves are greater than 33 hertz, and can thus be considered "rigid." According to the DC-SE-005-SD, the natural frequency of the product liquid sampling autoclave is 14 hertz and is thus "flexible."

The NRC staff, on the basis of its review of the SAR and these two design calculations, finds no justification for using an equivalent static load methodology. The NRC staff also notes that for the product blending and feed autoclaves, the applicant has not used a 1.5 factor for the peak horizontal ground accelerations as recommended by NUREG-0800 (NRC, 1987a) Section 3.7.2 (b.iii) "Equivalent Static Load Method."

In the absence of proper justification for using or not using an equivalent static load method, the NRC staff has applied the 1.5 factor across the entire range of three autoclave types and has presented the results, along with the applicant's results, in Table 4.13. This table shows that the margin of safety is reduced by using the more conservative approach suggested in NUREG-0800. However, the margins are still positive for the feed and blending autoclaves and are thus acceptable.

The applicant's position on the seismic stability of the product liquid sampling autoclave is not so well developed as that of the other two autoclave types because the sampling system is not rigid. The applicant submitted two evaluations of the proposed product liquid sampling autoclave. They are reports DC-NT-IIC2-AA and 04539001.

The product liquid sampling autoclave is supported on a foundation by a method which the applicant has not completely described at this time. The tilting operation is accomplished by means of two hydraulic piston/cylinder assemblies and scissors mechanisms which apply vertical and longitudinal force components to pins attached to the rear of the autoclave. The tilting mechanism is also not completely described at this time. Consequently, the NRC staff has not reviewed the design for support of the product sampling autoclave, including its tilting mechanism.

Table 4.13 Determination of appropriate safety class category for CEC autoclaves based on resistance to overturning and sliding during a design basis earthquake

Component	Margin of Safety Against Overturning		Margin of Safety Against Sliding		Conclusion
	LES	NRC	LES	NRC	
Feed Autoclave UF_6 (48Y) Cyl.	5.93 6.49	3.93 4.33	6.08 6.08	3.96 3.97	Stable against seismic motion
Product Blending Autoclave UF_6 (30B) Cyl.	6.23 7.7	4.14 4.9	6.09 6.11	3.96 3.96	Stable against seismic motion
Product Liquid Sampling Autoclave Down Position	NA	NA	NA	NA	NRC staff's review of specification basis for acceptance
Sampling Autoclave 30°-Tilted Position	NA	NA	NA	NA	NRC staff's review of specification basis for acceptance
UF_6 Cylinder in Cradle	6.08	3.96	Mechanical stop designed for 1.54 Kip	Stop should be designed for 2.16 Kip	NRC staff's review of specification basis for acceptance

In the evaluations which are available, the applicant specifies that seismic forces do not act concurrently; the NRC staff does not agree. (See Regulatory Guide 1.92 for appropriate methods of combining three components of earthquake motion.) The NRC staff notes that the dimensions used in the Design Calculation DC-NT-IIC2-AA do not correspond with the dimensions in SAR Figure 5.2-29. The applicant has stated that the strut or hydraulic cylinder is not designed for tension but does not explain how seismic stability is ensured if the strut is loaded in tension.

The analysis to check the stability of the UF_6 cylinder in the cradle is similarly lacking because of the following: (1) three orthogonal seismic components were not considered, (2) no factor of 1.5 was used as a part of the justification for simplified analysis, and (3) the UF_6-30B cylinder is inside the flexible autoclave and thus requires the higher acceleration of the entire autoclave, that is, 0.128 g vertical and 0.12 g horizontal. The NRC staff has applied these factors to the proposed design and concludes that the UF_6 cylinder is adequately protected against overturning, but that the mechanical stop to prevent sliding is under-designed. The NRC staff believes that the mechanical stops to prevent sliding of the UF_6 cylinder in the 30°-tilted position should be designed for 2.16 kips instead of 1.54 kips cited in the design calculation.

The NRC staff recognizes that the SAR description of the sampling autoclave represents a preliminary design, and that it may not be realistic to expect all design aspects to have a complete supporting analysis. Furthermore, the NRC staff also recognizes that it is possible to design a sampling autoclave and tilting mechanism which is adequate for the DBE.

After this NRC staff review of the DBE response of the sampling autoclave, the applicant submitted design specifications (SP-539000-40-3) for the product liquid sampling autoclave (LES, 1993h). The NRC staff reviewed this design specification and concludes that a product liquid sampling autoclave built to the specification would resolve the deficiencies discussed above. These deficiencies include: (1) lack of a complete design for support of the autoclave, (2) lack of a complete design for the tilting mechanism, and (3) inadequate mechanical stop to prevent the UF_6 cylinder from sliding while in the tilted position.

Thus, in order for the NRC staff to confirm that the final design of the product liquid sampling autoclave is in accord with the specification (SP-539000-40-3) reviewed in the SER, the NRC staff proposes the following license condition:

- As an element of the required preoperational inspection process, the applicant will supply materials described in FDI Specification SP-539000-40-3 to the NRC for review and approval. The materials, identified in Section 1.6 of the specification, shall include:

  A. Drawings, including dimension drawings and hydraulic connection drawings

  B. Technical data, including: (1) design calculations, (2) descriptive literature, and (3) material certifications

  C. ASME Code documents and special requirements, including: (1) ASME forms in accordance with Section VIII, Division 1, (2) hydraulic test results, (3) photograph of nameplate, and (4) the seller's QA plan consistent with 10 CFR Part 50, Appendix B and NQA-1.

- Items A and B(1) must be the final delivered design and be complete in sufficient detail to permit a second party review.

4.7.2 Autoclave Foundation Analysis

The foundation analysis discussed in Design Calculation DC-SE-005-SD (LES, 1993c) consists of an analysis of the fundamental frequency of each of the three autoclave types, a seismic analysis for the stability and soil-bearing capabilities of the foundations, and a foundations stress analysis and design. The design calculation discussed here includes only the concrete design; design of the autoclave supports is considered in the autoclave mechanical component analysis.

The feed and blending types of autoclaves are modeled as simple pipes supported on two saddles at each end. The model was a "stick" model appropriate for determining natural frequency. The product liquid sampling autoclave was modeled as a simple pipe supported by the tilting linkage. The NRC staff finds that methods used by the applicant are appropriate for determination of the natural frequency of the autoclaves.

The applicant uses the Southern Building Code Congress International, Standard Building Code, an approach acceptable to the NRC staff. SER Table 4.3 and 4.4 indicate the actual load combinations required; however, the only loads of importance are those of dead weight and seismic load, and the NRC staff finds that the load combinations used are acceptable.

The NRC staff accepts the analysis performed for the autoclave foundations. Minor errors identified in the applicant's analysis have no impact on the design.

The NRC staff reviewed applicant's analysis regarding overturning or sliding of the foundation and finds the design acceptable. The results of this calculation are summarized in Table 4.5.

4.7.3 Class I Electrical and Control Systems

SAR Table 4.6.1 identifies Class I control systems for the three autoclave types. Design calculation DC-SE-0008-SD (LES, 1993f) presents the analysis of these systems, which must function during the DBE. Each type of autoclave has air temperature protection elements, air pressure protection elements, control panels, and safety control panel supports. All of these control panels are located on the ground floor of the Separations Building.

The analysis performed by the applicant in DC-SE-0008-SD considered the seismic response of the control panels, including relays, the pressure transducer mount, and temperature elements, and pressure transducers. For the temperature elements and pressure transducers, the applicant stated that because there are no moving parts in the temperature or pressure sensors, there is nothing to excite during an earthquake. It was stated that as long as the cables and mounts are seismically qualified, the temperature elements and pressure transducers will be adequately protected. The NRC staff concurs with the applicant.

<u>Control Panels</u>

The NRC staff found the applicant's analysis performed to show that the control panels are suitably qualified for seismic motion to be deficient. First, the moment to overturn the electrical panel enclosure uses a 36-inch moment arm rather than a 42-inch moment arm from the floor to the center of gravity, with a resulting small overturning moment. Second, the resisting moment uses an incorrect and non-conservative moment arm of 8 inches instead of 4 inches, with a resulting large resisting moment. Third, because the applicant made no estimate of the natural frequency of the system, the applicant's assumption that the maximum seismic load is 0.099g is appropriate. Although conservative, this assumption is not correct for overturning.

The NRC staff determined that the natural frequency of the cabinet is on the order of 211 HZ, well above the 33 Hz considered by the applicant. Thus, the maximum horizontal acceleration causing overturning is 0.046g, which corresponds to "rigid system" acceleration according to the LES DBE Spectra. After making the above noted corrections, the NRC staff determined that the cabinet will have a 21 percent margin against overturning and is thus acceptable.

Pressure Transducer Mount

The applicant provided analysis to show that the pressure transducer mount is suitable for the design basis seismic loading. The NRC staff assumed the transducer was mounted flexibly because the applicant used a mid-field input of 0.099g instead of the 0.046g for a rigid body. The NRC staff confirmed that this value is appropriate in estimating the natural frequency of the mounting system to be 37 Hz, which is close to the 33 Hz control point A (Reference Regulatory Guide 1.60), so that the system may be considered flexible.

The NRC staff checked seismic loads for the transducer mount and found them to be conservative. The NRC staff also checked the stress analysis and found it to be well under the stress allowable as permitted by the manual of steel construction (Ninth Edition) of the AISC code.

The applicant states that autoclave heaters will be shut off in the event of relay chatter, which is not eliminated in the proposed design. However, according to the applicant, should relay chatter occur, no overheating of the autoclave results and, the relays must be manually reset by an operator. The NRC staff concludes that relay chatter will not affect the fail-safe operation of the heaters and is thus acceptable.

4.8 Confinement Barriers and Systems

The primary confinement barriers used at the CEC include cylinders for feed, product, and tails material, process piping, and desublimer tubes. Secondary confinement for UF_6 in the liquid state is provided by feed, blending, and sampling autoclaves. The analysis of systems, structures, and components presented in SAR Section 4.2 does not identify these components as important to safety in CEC process operations because either other systems provide the required safety functions or component failure under design basis conditions does not lead to a release exceeding the NUREG-1391 criteria. Because ANPR guidance is that design requirements be commensurate with safety function, the above barriers are reviewed for acceptability against chemical process construction standards.

Feed, product, and tails cylinders are fabricated and tested in accordance with ASME Boiler and Pressure Vessel Code Section VIII (ASME, 1989c) and the ANSI N14.1 standard for UF_6 packaging (ANSI, 1990). Specifications include material of construction, wall thickness, volume, hydrostatic test pressure, and allowable leak rates. The autoclaves are also fabricated in accordance with ASME Code Section VIII. CEC process piping and desublimers are fabricated of aluminum alloy in accordance with ANSI/ASME B31.3, Chemical Plant and Petroleum Refinery Piping. Specifications include material of construction, operating conditions, and wall thickness. The NRC staff concludes that equipment fabricated and tested in accordance with these standards are adequate to protect public health and safety at the CEC.

4.9 Ventilation Systems

The CEC uses thirteen heating, ventilation, and air conditioning (HVAC) systems and one process gas effluent system, the Gaseous Effluent Vent System (GEVS) in order to provide general ventilation and conditioning of gaseous effluents. Of these fourteen systems, only the GEVS and a portion of the Technical Services Area (TSA) HVAC system are intended to handle potentially contaminated air. Under normal operating conditions, the remaining twelve systems handle air which is not contaminated; under accident conditions, in which contamination could be present, the systems are designed to shut down. As a consequence of this design only the GEVS and the TSA HVAC system have a role in protecting worker or public health and safety beyond maintaining acceptable environmental conditions under normal operating circumstances. This section describes the GEVS and the TSA HVAC systems, identifies the role of these systems in protecting public health and safety, and evaluates the designs against the ANPR requirements. The ANPR requirements for ventilation systems are:

- The desired air flow direction shall be maintained under normal operating and accident conditions
- The system shall accommodate changes in operating conditions and be capable of controlling off-gases
- The continuity of necessary ventilation shall be maintained
- Provision shall be made for testing
- The systems shall be designed to permit continued occupancy of areas needed for normal operations, safe shutdown, and maintaining safe shutdown
- The systems shall be designed to confine hazardous materials during normal operations.

<u>Gaseous Effluent Vent System</u>

The GEVS is a once-through conditioning system designed to remove uranium compounds and HF from CEC process system effluents. The system is hard-piped to the air space of feed, blending, and sampling autoclaves; the feed purification, product, and blending vent desublimer discharge lines; and the discharge lines of mobile vacuum pump sets. The system is connected by means of elephant trunks (lengths of flexible tubing) to areas where normal connecting of piping could release small quantities of contaminated vapor. These areas includes the autoclaves, the cylinder stations, the product and tails vacuum pumps, and the cascade sampling points. The GEVS also receives air from TSA hoods and, in the unlikely event of activation, from the Contingency Dump System. The GEVS is comprised of a series arrangement of pre-filters, High Efficiency particulate Air (HEPA) filters, activated carbon filters, and exhaust fan. The pre-filters, HEPA filters, and carbon beds are each configured as a bank of five parallel units, with four on line at a time and one retained as a spare. The removal efficiency of the pre-filters is rated at 99.7 percent for particles of a diameter greater than five microns. The removal efficiency of the HEPA filters is rated at 99.97 percent for particles of 0.3 micron diameter. The removal efficiency of the carbon filters for HF is rated at 99 percent. Flow through any single filter is 0.47 m^3/s (1,000 cfm), and filter pressure drop is monitored to indicate the need for replacement. The fan throughput is 1.84 m^3/s (3,900 cfm) at the design pressure drop, and a spare fan is provided. System pressure is monitored and air fan speed adjusted to maintain required flow and design pressure. The system operates at subatmospheric pressure, and HF monitors and alarms are installed upstream of the filters and the stack in order to alert operators to off-normal conditions. The system is designed in accordance with ANSI N509-1990.

The series design of the system with exhaust fan meets the ANPR requirement that flow direction be maintained. Using pressure monitors with a variable-speed fan meets the requirement that the system accommodate changes in conditions, and providing spares for individual components meets the requirement for continuity of operation. The system is continuously monitored, testable, and designed to remove hazardous components by using filters and adsorption beds, and thereby helps maintain safe conditions to operate or shutdown the facility. The NRC staff concludes that the GEVS meets the requirements of the ANPR and is adequate for protection of worker and public health and safety.

Technical Services Area HVAC System

Decontamination of equipment, analysis of UF_6 samples, and storage of waste are activities conducted in the TSA which could contaminate ventilation air. The primary confinement of contamination in the TSA is achieved by use of hoods and confined work areas vented to the GEVS. Additional activities conducted in the TSA, including chemical analysis support, electrical and mechanical maintenance, and material storage do not involve the potential for development of airborne contamination. To meet the ventilation needs of the TSA, a HVAC system with a single intake and two parallel exhaust paths is proposed. Fresh outside air is drawn into the system by three parallel fans through three parallel air handling units. Each air handling unit is comprised of two filters, and heating and cooling coils. The filters are designed to remove atmospheric dust and not to filter contaminated air. The cooling and heating coils maintain acceptable environmental conditions in the ventilated workspace. The combined exhaust of the three supply units is split into a ventilation air stream for areas with no potential for airborne contamination and a ventilation air stream for the potentially contaminated areas. After passing through clean areas, the non-contaminated air is drawn into parallel fans, and the major portion is recycled to the supply unit. Ten percent of the clean exhaust fan stream is released to the atmosphere through the plant stack. The air stream which passes through the potentially contaminated work areas is drawn through three parallel filter/fan units. Each of the units is comprised of a pre-filter, a HEPA filter, and an exhaust fan. The combined exhaust of the three fans is released directly to the plant stack with no recycle to the supply unit. Each of the three filtration exhaust fans is rated at 33 percent of capacity at design pressure drop, and fan differential pressure is monitored. The system is designed in accord with ANSI N509-1990.

The TSA HVAC System does not service areas with potential for release of contamination at levels with safety significance and does not interact with systems serving the control room or areas with significant UF_6 inventory. The series design of the system serving potentially contaminated areas does maintain desired flow direction. The system is monitored, testable, and designed to confine hazardous material which could be present. On the basis of the above description and analysis, the NRC staff concludes that the TSA HVAC System meets the requirements of the ANPR and is adequate for protection of worker and public health and safety.

5 INSTRUMENTATION AND CONTROLS

Instrumentation and controls at the Claiborne Enrichment Center (CEC) support continuous operation of the separation process, and worker and public health and safety. In this evaluation, instruments with a primary function to monitor or direct operation are termed process controls, and systems with a primary function to prevent releases, facilitate process shutdown, or protect worker and public health and safety are termed protection systems. Among the most significant protection systems are those which are designated as important to safety. The objective of this evaluation of CEC instrumentation and control systems is to establish that system design and operation provide acceptable protection of worker and public health and safety under normal and off-normal operating conditions. The review is based on information provided in the applicant's Safety Analysis Report (SAR) (LES, 1993a) and in response to Requests for Additional Information (RAI) (LES, 1992a, 1992b, and 1992c).

The Advance Notice of Proposed Rulemaking (ANPR) (NRC, 1988a) for regulating uranium enrichment facilities provides guidance on the acceptability of instrumentation and control systems. The ANPR requires that instrumentation and control systems be provided to monitor variables and operating systems important to safety during normal and off-normal operation, accident conditions, and shutdown situations. In addition, the overall confinement system, confinement barriers, and other systems which affect plant safety must be continuously monitored. Controls must maintain variables within prescribed operating ranges under all normal conditions and control systems must be designed to fail to a safe state. The ANPR specifies the protection systems shall be designed to:

- Initiate actions to ensure that design limits are not exceeded
- Sense potentially hazardous conditions
- Have reliability and testability
- Maintain function with loss of a single active component
- Maintain function with removal from service of any component
- Fail to a safe state in loss of power.

This section briefly describes the CEC process control and protection systems, summarizes limiting conditions of operation, and evaluates the proposed design in light of the ANPR guidance. The equipment and instrumentation descriptions presented in this chapter are drawn from the CEC SAR and the analyses and evaluations are independent NRC staff analyses and evaluations.

5.1 Summary Description

This sub-section describes control systems in the order of the movement or flow of uranium hexafluoride (UF_6) through the separation process. The Safety Evaluation Report (SER) Section 3 has more detailed descriptions of the equipment; CEC SAR Section 6.8 has detailed flow diagrams showing the location of instruments, controls, and controlled elements.

The overall CEC control strategy is to locate process and safety control functions at the local equipment level, in Local Control Centers (LCC), with the upper-level Central Control Room (CCR) functions for monitoring and reviewing operations, integrating individual sub-system operation, and centralizing responses to off-normal conditions. CEC instrumentation and controls include administrative and operating procedures, automatic electromechanical systems, and operator-activated switches. Many systems--feed autoclaves, feed purification and vent, separation cascades, product high-pressure pumps and cylinders, product vent, tails high-pressure pumps and cylinders, product blending autoclave, product blending cylinders and vent, product sampling autoclaves, and contingency dumps--have LCCs. Each LCC can function without CCR intervention; however, all information available in the LCC is duplicated at the CCR level, and actions which can be taken by the LCC can be taken by the CCR.

5.1.1 Cylinder Receipt and Handling

Operating procedures and administrative controls for receiving and handling cylinders protect against UF_6 releases by preventing unsafe operations such as heating of over-filled cylinders. Cylinders are received from and shipped offsite through the Cylinder Receipt and Dispatch Building (CRDB), which serves as an inventory location. This building receives, inspects, and weighs full feed cylinders, and empty product and tails cylinders. It also pressure-tests empty product and tails cylinders for integrity against damage or leaks. Feed cylinders transferred to the Separations Building are weighed a second time in the Cylinder Handing Area before they are placed in the feed autoclaves. The weigh station has a reader and printout facility.

5.1.2 Feed Systems Controls

CEC feed systems remove light components from the feed material and provide continuous flow of UF_6 to the separation cascades. These functions are duplicated in each of the three CEC plant units. The major elements of the feed system are autoclaves, desublimers, purification cubicles, and associated piping and valves. The primary process control circuits used in the purification and feed operations are represented schematically in Figure 5.1. The figure is a simplification of the actual system and does not identify signal paths for all process control and protection circuits. Coordinated, upper-level control of the purification and feed operations is provided by state switches on the autoclave, desublimer, and purification cubicle control panels. Systems activated by using the autoclave, desublimer, or purification cubicle state switches are specific to the function performed in that state. Eight states are defined for the autoclave, six states are defined for the desublimer, and four states are defined for the purification cubicle. Operator selection of a state opens or closes valves (the switch position generally remains unchanged during the particular operation), activates control systems and interlocks, and activates or inhibits appropriate protection systems.

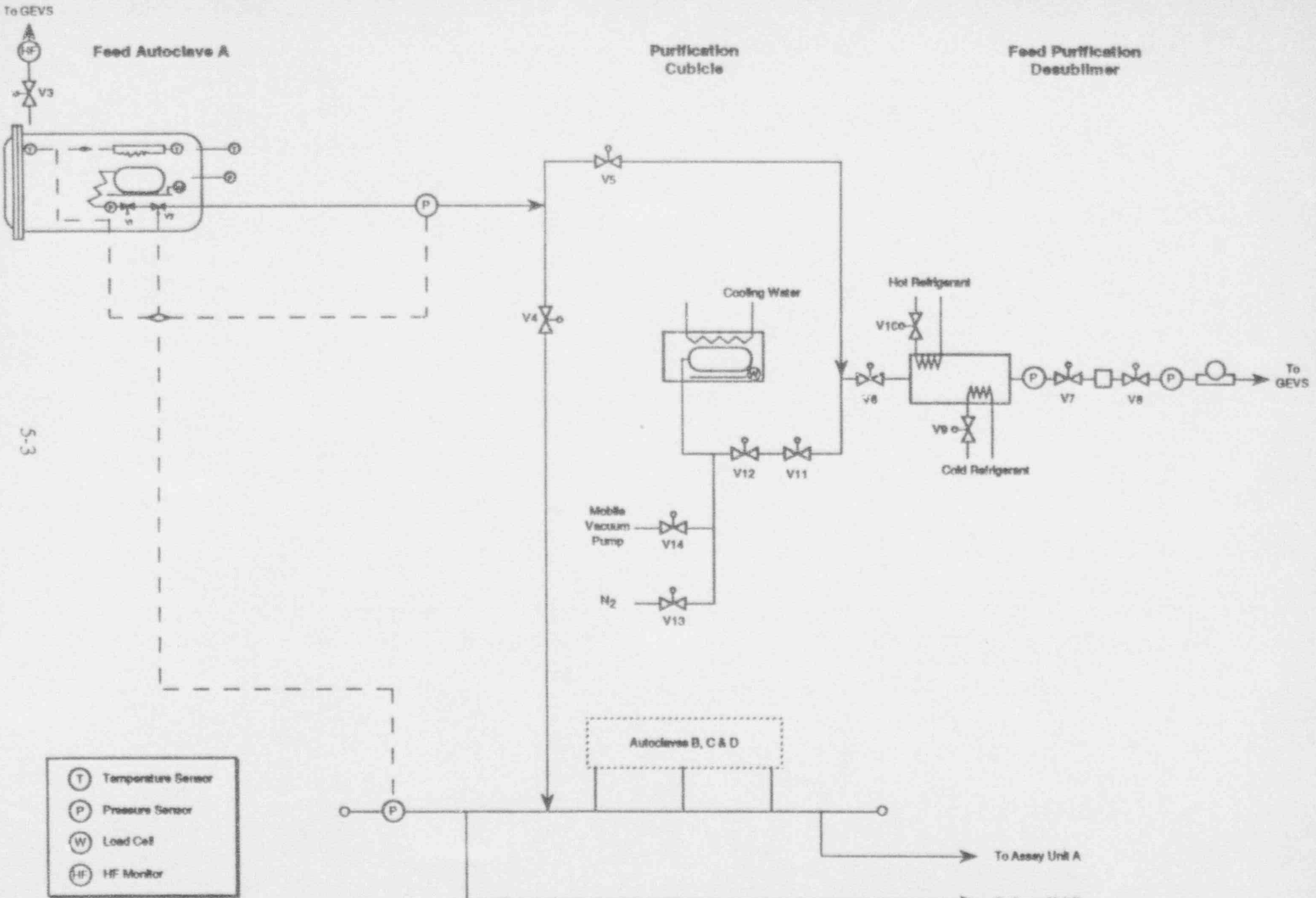

Figure 5.1 Schematic for feed system instrumentation and controls

Equipment states and associated representative valve positions are summarized in Table 5.1 for the feed autoclave, in Table 5.2 for the feed purification desublimer, and in Table 5.3 for the purification cubicle. Valve identification is referenced to the schematic diagram of Figure 5.1.

Table 5.1 Feed autoclave states and valve positions

State	Valve Position			
	V1	V2	V4	V5
Isolate	closed	closed	closed	closed
Cold Purify	open	open	closed	open
Heat-up	closed	closed	closed	closed
Hot Purify	open	open	closed	open
Standby, manual	closed	closed	open	closed
Standby, auto	closed	closed	open	closed
On-line	open	open	open	closed
Heels Removal	open	open	closed	open

Table 5.2 Feed purification desublimer states and valve positions

State	Valve Position				
	V6	V7	V8	V9	V10
Heat	closed	closed	closed	closed	open
Gas-over	open	closed	closed	closed	open
Chill	closed	closed	closed	open	closed
Standby	closed	open	open	open	closed
On-line	open	open	open	open	closed
Purification	open	closed	closed	open	closed

Table 5.3 Feed purification cubicle states and valve positions

State	Valve Position			
	V11	V12	V13	V14
Isolate	closed	closed	closed	closed
On-line	open	open	closed	closed
Vacuum	closed	closed	closed	open
Nitrogen	closed	closed	open	closed

The primary process control functions of the autoclave are heating and evaporating the UF_6 and controlling the flow leaving the autoclave. Control of heating rate is determined by monitoring the cylinder exit pressure, which is directly related to UF_6 temperature and phase state. Flow rate of UF_6 is controlled by monitoring autoclave exit line pressure, which is determined by the pressure drop across the control valve located inside the autoclave. Positioning of valves other than the autoclave exit flow control valve is determined by the states selected on the autoclave and desublimer state switches. Thus, for cold purification, the autoclave state switch is in the cold purify position, the desublimer state switch is in the purification position, the valves leading to the cascades and the purification cubicle are closed, the inlet valve to the desublimer is open, and cold refrigerant cools the desublimer. Similar considerations and valve positioning are determined for the other purification and feed functions.

The primary protection function of the purification and feed control systems is prevention of UF_6 release from the process equipment. Release mechanisms include breaking the confinement barriers by external force or internal expansion of the UF_6. Internal threat to the confinement barriers is indicated by elevated temperature or pressure, or by excess UF_6 inventory. Instrumentation and controls are provided to monitor variables which indicate threat to the confinement barriers, and to prevent occurrence of a potentially dangerous situation or to assist mitigative action. Protective controls for the feed autoclave and purification equipment are presented in Tables 5.4 and 5.5, respectively. In addition, a hydrogen fluoride (HF) monitor located in the Gaseous Effluent Vent System (GEVS) vent line is used to check for leaks prior to opening an autoclave. Set points for response of the feed autoclave and feed purification protection and control systems are:

- Autoclave air temperature of 127 °C (260 °F)
- Autoclave air pressure of 170,300 pascals (24.7 psia)
- Feed cylinder pressure of 344,740 pascals (50 psia)
- Heater element surface temperature of 150 °C (302 °F)
- Autoclave exit line pressure of 8,000 pascals (1.16 psia)
- Desublimer tube pressure of 5,000 pascals (0.725 psia).

Table 5.4 Feed autoclave protective controls

Function	Monitored Variable
De-energize heaters	• high autoclave pressure, redundant • high autoclave temperature, redundant • high heater element temperature • high cylinder exit line pressure • incorrect position of autoclave exit line valves
Terminate flow to plant unit header	• high autoclave exit line pressure, redundant
Isolate from GEVS	• position of GEVS vent line valve
Prevent door opening	• position of GEVS vent line valve • high autoclave air pressure
Alarm on excess cylinder weight	• autoclave load cell

Table 5.5 Feed purification desublimer protective controls

Function	Monitored Variable
Terminate inlet flow	• high desublimer pressure
Terminate outlet flow	• low desublimer pressure • high vacuum pump inlet pressure
Alarm on excess cylinder weight	• purification cubicle load cell

5.1.3 Separation Cascades Controls

Seven cascades are grouped into each of the two assay units which comprise each of the three CEC plant units. Identical process control and protection systems are provided for each of the cascades.

Cascade process control systems provide uniform feed and exit flow of UF_6 and maintain centrifuge operating temperature. To ensure proper feed rate, each cascade receives feed through a control system comprised of a pressure sensor, resistor orifice, and control valve. To provide a constant flow through the orifice, the control valve position is adjusted in response to the pressure signal. A similar arrangement is used to control product take-off flow rate, and a pressure control device maintains the tails terminal pressure at the required value. UF_6 control valves are powered by a 24-volt direct current system which has battery back-up. Cascade temperature is controlled by a closed-loop cooling water system designed to maintain centrifuge inlet temperature between 30 and 32 °C (86.0 and 89.6 °F). The temperature of the demineralized water used in the closed loop is monitored and adjusted by indirect exchange with cold water from the main plant cooling water system. The temperature of the cooling water entering each cascade is monitored, and flow is adjusted to remove the required heat.

The cascade protection system measures cascade header pressure and current drawn by the centrifuges, and monitors the status of the feed, contingency dump, cooling water, and product and tails take-off systems. Valves at the product and tails take-off points can isolate the cascade, and automatic control valves dump the cascade contents to the product or tails cylinders or to the contingency dump system. Centrifuge rotor failure, or "crash," is detected by monitoring the centrifuge current. Rotor failure causes increased pressure in the individual centrifuge, which, in turn, activates an isolation device for each machine.

5.1.4 Product Take-Off Controls

Systems provided for product take-off include low- and high-pressure vacuum pumps, cylinder stations, and desublimer venting equipment. The compression, desublimation, and venting functions of this equipment each have process control and protection systems. The arrangement of instruments and controls provided for the vacuum pump systems is represented in Figure 5.2. A state switch provided for the series low-pressure pump set has two allowed states: off, with pump set stopped, for maintenance; and on-line, with pump set running. During normal operation, the inlet and outlet valves of the pump set are open, and no additional process controls specific to the pumps are used. State switches for each of the two parallel high-pressure pumps have four allowed states: on-line (pump running), maintenance (pump stopped), vent (pumped stopped) and warm-up (pump running). During normal operation, the inlet and outlet valves of the pumps are open, the bypass valve is closed, and no additional process controls specific to the pumps are used. Pump states and representative valve positions are summarized in Table 5.6.

Table 5.6 Product take-off pump states and valve positions

State	Valve Position				
	V1	V2	V3/V6	V4/V7	V5/V8
LP Pump					
on-line	open	open	-	-	-
maintenance	closed	closed	-	-	-
HP Pump					
on-line	-	-	open	open	closed
maintenance	-	-	closed	closed	open
vent	-	-	open	closed	open
warm-up	-	-	closed	closed	open

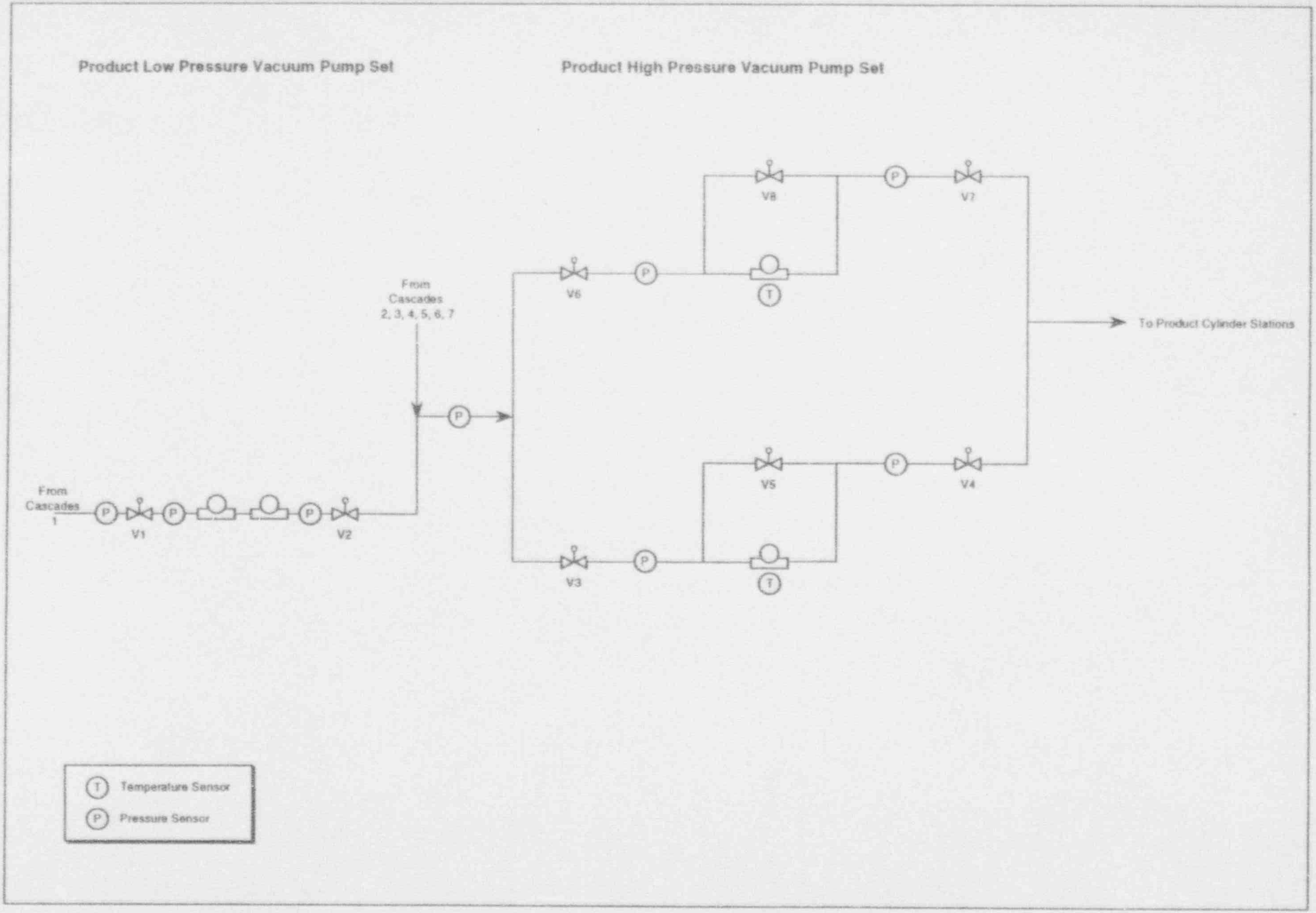

Figure 5.2 Schematic for product vacuum pump instrumentation and controls

Protection controls for the low-pressure vacuum pump set include pressure monitors and isolation valves. A sequenced set of high, upstream pressures initiates a dump to the tails take-off system, closes the pump set inlet valve, closes the outlet valve, and stops the pumps. High pressure detected downstream from the low-pressure pump set initiates the same actions. The actions are sequenced to the level of the elevated pressure and protect against high pressure in the dump mode and gross leakage. The detection of high pressure in the header handling the combined flow of all cascades in the assay unit initiates a tails dump of all cascades in the assay unit.

Protection systems for the high-pressure vacuum pumps include gauges measuring inlet and outlet pressure and pump flange temperature for each pump. The first indication of high inlet pressure closes the inlet and outlet valves, opens the by-pass valve, and runs the pump in a recycle mode. Continued increase of inlet pressure closes inlet and outlet valves and stops pumps for both the low- and high-pressure pumps sets. These actions protect against gross leakage and UF_6 desublimation. An indication of high pressure on the high-pressure pump outlet closes the inlet and outlet valves, opens the by-pass valve, and runs the pump in a recycle mode. An indication of elevated or depressed pump flange temperature causes an alarm which protects against pump overload or UF_6 desublimation.

Enriched UF_6 is cooled and desublimed into product cylinders through indirect contact with air in product cylinder stations. A product vent desublimer system removes light gas components which concentrate in the product stream. The equipment arrangement is represented in Figure 5.3. Process control of the cylinder station and desublimer is effected through operator manipulation of state switches, which automatically establish valving patterns required for the selected operation. The cylinder station state switch establishes on-line, auto-vent, vent, maintenance, and isolate as allowed states. The desublimer state switch establishes heat, gas-over, chill, standby, on-line, and auto-vent as allowed states. For example, operator selection of "on-line" on the cylinder station state switch opens the inlet header valve and closes the desublimer inlet valve, and thereby directs UF_6 product into the cylinder station and isolates the desublimer from the inlet flow. Air recirculated to the cylinder stations is cooled by indirect contact with cooling water supplied by the spray cooling water system. Cooling water pressure and temperature are monitored for control of supply rate and temperature. The pressures and temperatures of the cold and hot refrigerant are monitored to control supply flow rate and temperature. States and representative valve positions for the product cylinder station and the product vent desublimer are presented in Tables 5.7 and 5.8, respectively.

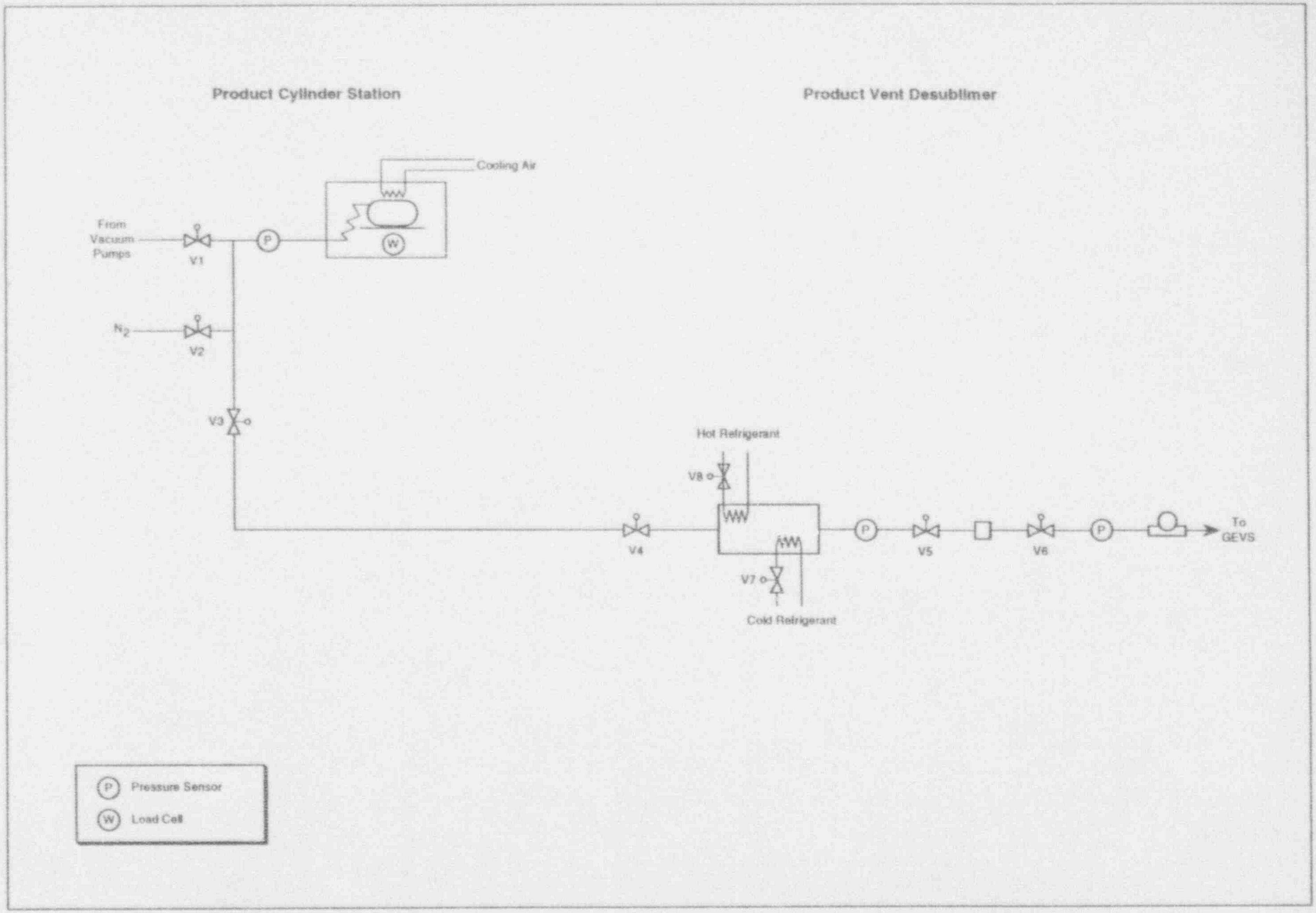

Figure 5.3 Schematic of product take-off cylinder station instrumentation and controls

Table 5.7 Product cylinder station states and valve positions

State	Valve Position		
	V1	V2	V3
On-line	open	closed	closed
Auto-vent	closed	closed	open
Vent	closed	closed	**
Maintenance	closed	open	closed
Isolate	closed	closed	closed

** Controlled by desublimer state switch

Table 5.8 Product vent desublimer states and valve positions

State	Valve Position				
	V4	V5	V6	V7	V8
Heat	closed	closed	closed	closed	open
Gas-over	open	closed	closed	closed	open
Chill	closed	closed	closed	open	closed
Standby	closed	open	open	open	closed
On-line	open	open	open	open	closed
Auto-vent	open	closed	closed	open	closed

Protection systems for the product cylinder station isolate the cylinder on signals of elevated cylinder pressure or excess cylinder weight. Protection systems for the desublimer close the inlet valve on high desublimer pressure and the outlet valve on low desublimer pressure or high vacuum pump inlet pressure. Set points for response of the product take-off protection and control systems are:

- Primary header pressure of 830 pascals (0.12 psia)
- Low-pressure pump inlet pressure of 830 pascals (0.12 psia)
- Low-pressure pump outlet pressure of 9,030 pascals (1.31 psia)
- Secondary header pressure of 12,000 pascals (1.74 psia)
- High-pressure pump inlet pressure of 9,030 pascals (1.31 psia)
- High-pressure pump outlet pressure of 50,000 pascals (7.25 psia)
- High-pressure pump flange temperature of 45 °C (113 °F)

- Product cylinder inlet pressure of 80,000 pascals (11.6 psia)
- Desublimer tube pressure of 5,000 pascals (0.725 psia)
- Desublimer vent line vacuum pump inlet pressure 800 pascals (0.116 psia).

5.1.5 Tails Take-Off Controls

The tails take-off system compresses the UF_6 from cascade pressure to tails cylinder pressure and desublimates the UF_6 into the tails cylinders. A feed purification vent system occasionally vents a tails cylinder. Vacuum pump compression and spray cooling water desublimation equipment have process control and protection systems. The arrangement of instruments and controls provided for the vacuum pump systems is represented in Figure 5.4. A state switch for the series low-pressure pump set has two allowed states: off, with pump set stopped, for maintenance; and on-line, with pump set running. During normal operation, the inlet and outlet valves are open, and no additional process controls specific to the pumps are used. State switches for each of the three parallel high-pressure pumps have four allowed states: on-line (pump running), maintenance (pump stopped), vent (pumped stopped) and warm-up (pump running). During normal operation, the inlet and outlet valves open, the bypass valve is closed, and no additional process controls specific to the pumps are used. Pump states and representative valve positions are summarized in Table 5.9.

Protection controls for the low-pressure vacuum pump set have pressure monitors and isolation valves. A sequenced set of high upstream pressures initiates a dump to the product take-off system, closes the pump set inlet valve, closes the outlet valve, and stops the pumps. High pressure detected downstream from the low-pressure pump set initiates the same actions. The actions are sequenced to the level of the elevated pressure and protect against high pressure in the dump mode and gross leakage. Detection of high pressure in the header handling the flow of all cascades in the assay unit initiates a product dump of all cascades in the assay unit.

Table 5.9 Tails take-off pump states and valve positions

State	Valve Position				
	V1	V2	V3/V6/V9	V4/V7/V10	V5/V8/V11
LP Pump					
on-line	open	open	-	-	-
maintenance	closed	closed	-	-	-
HP Pump					
on-line	-	-	open	open	closed
maintenance	-	-	closed	closed	open
vent	-	-	open	closed	open
warm-up	-	-	closed	closed	open

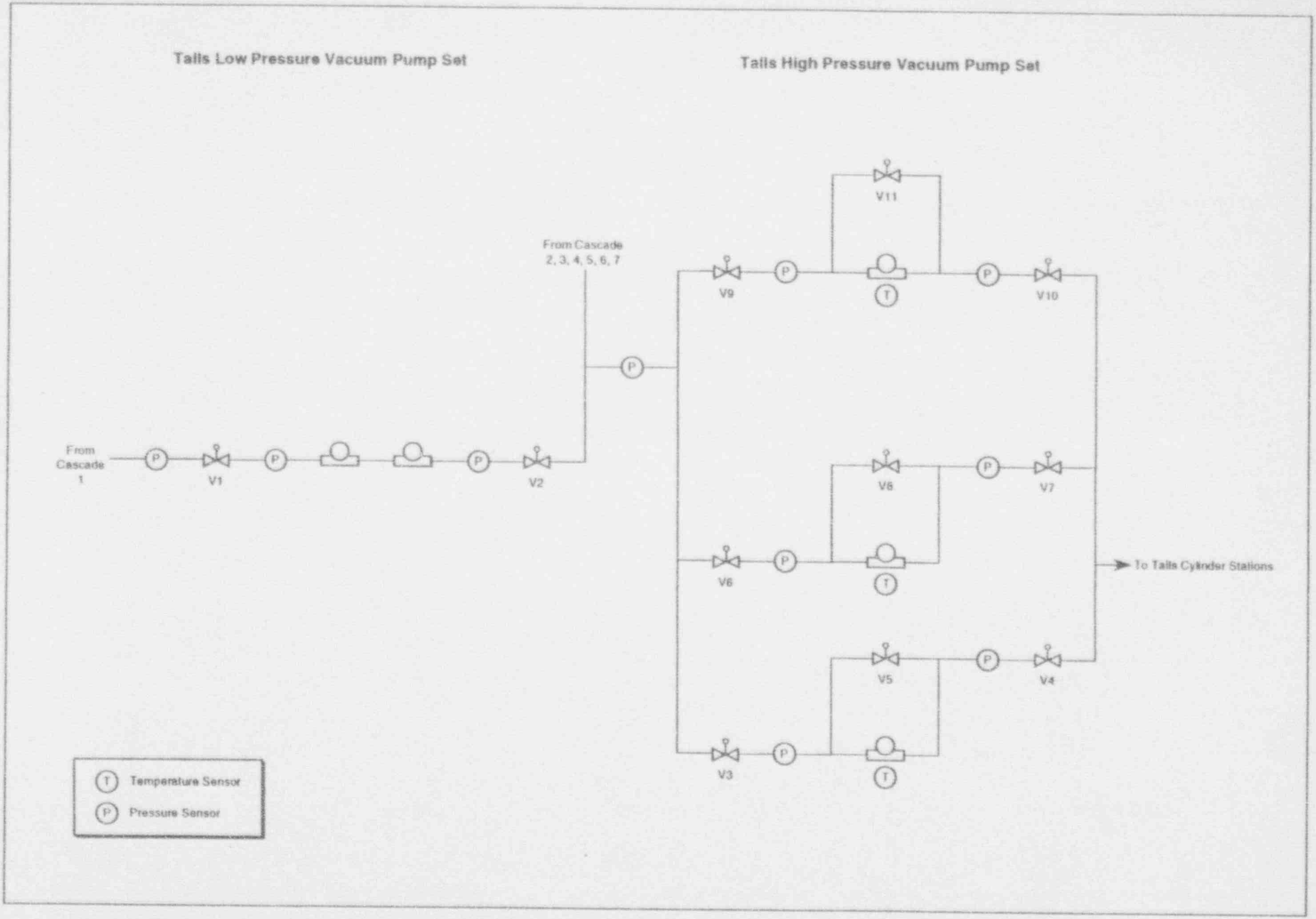

Figure 5.4 Schematic for tails vacuum pump instrumentation and controls

Protection systems for the high-pressure vacuum pumps include gauges measuring inlet and outlet pressure and pump flange temperature for each pump. The first indication of high inlet pressure closes the inlet and outlet valves, opens the by-pass valve, and runs the pump in a recycle mode. Continued increase of inlet pressure closes inlet and outlet valves, and stops pumps for both the low- and high-pressure pumps sets. These actions protect against gross leakage and UF_6 desublimation. An indication of high pressure on the high-pressure pump outlet closes the inlet and outlet valves, opens the by-pass valve, and runs the pump in a recycle mode. An indication of elevated or depressed pump flange temperature causes an alarm which protects against pump overload or UF_6 desublimation.

Depleted UF_6 is cooled and desublimed into tails cylinders by direct contact of the cylinders with water in tails cylinder stations. The equipment arrangement is represented in Figure 5.5. Process control of the cylinder stations is effected through operator manipulation of a state switch which automatically establishes valving patterns required for the selected operation. The state switch has four allowed states: on-line, vent, N_2 purge, and isolate. For example, operator selection of "on-line" on the cylinder station state switch opens the inlet header valve and closes the vent valve, and thereby directs UF_6 product into the two cylinder stations and isolates the feed purification desublimer system from the inlet flow. Cooling water supplied by the spray cooling water system is sprayed directly onto tails cylinders to desublimate the UF_6. Cooling water pressure and temperature are monitored for control of supply rate and temperature. Tails cylinder station states and representative valve positions are summarized in Table 5.10.

Table 5.10 Tails cylinder station states and valve positions

State	Valve Position		
	V1	V2	V3
On-line	open	closed	closed
Vent	closed	closed	open
N_2 purge	closed	open	closed
Isolate	closed	closed	closed

Protection systems for the tails cylinder stations isolate the cylinders on signals of elevated cylinder pressure or excess cylinder weight from either of the two stations serving an assay unit. Set points for response of the tails take-off protection and control systems are:

- Primary header pressure of 830 pascals (0.12 psia)
- Low-pressure pump inlet pressure of 830 pascals (0.12 psia)

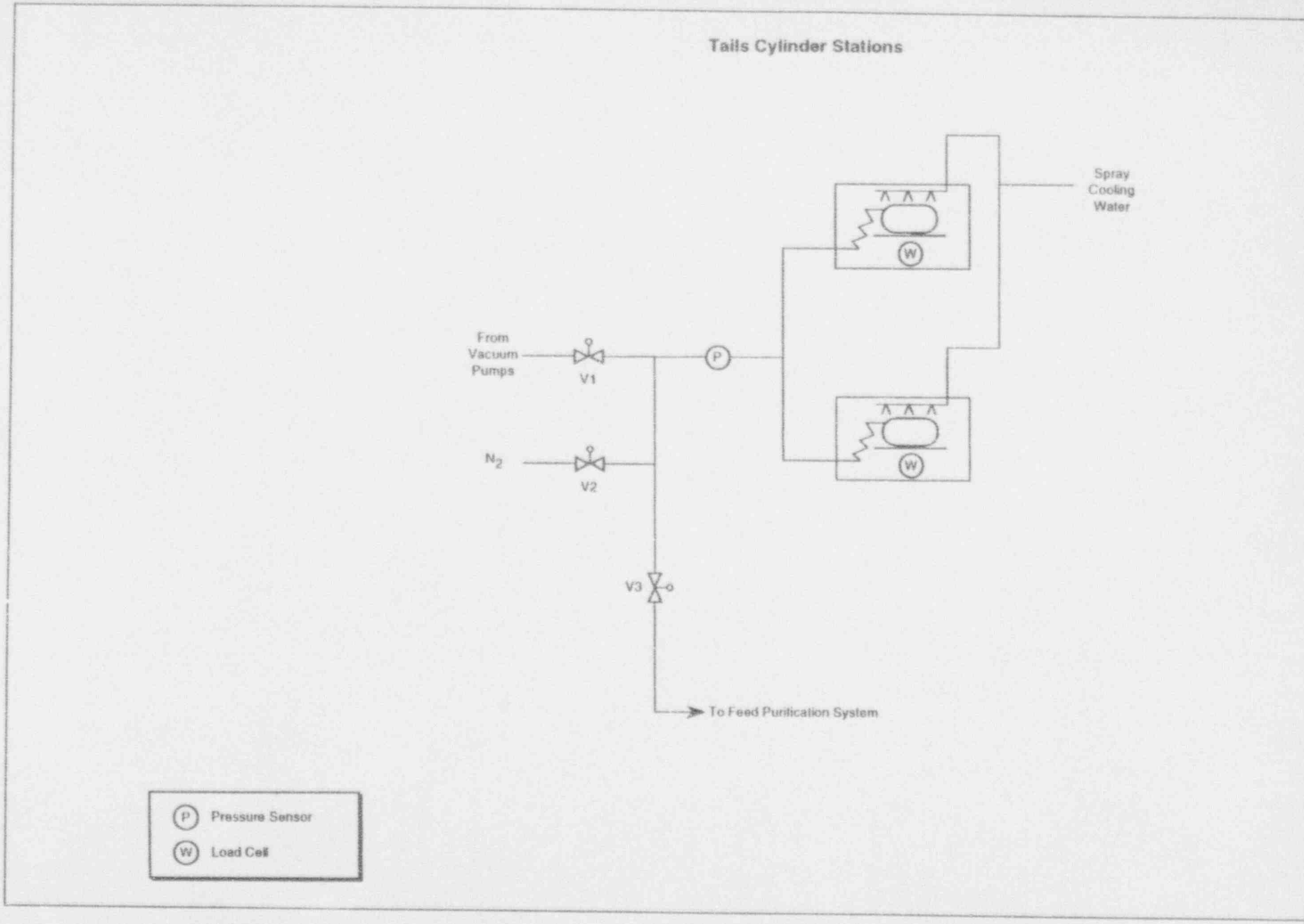

Figure 5.5 Schematic for tails take-off cylinder station instrumentation and controls

- Low-pressure pump outlet pressure of 9,030 pascals (1.31 psia)
- Secondary header pressure of 12,000 pascals (1.74 psia)
- High-pressure pump inlet pressure of 9,030 pascals (1.31 psia)
- High-pressure pump outlet pressure of 50,000 pascals (7.25 psia)
- High-pressure pump flange temperature of 45 °C (113 °F)
- Cylinder inlet pressure of 80,000 pascals (11.6 psia).

5.1.6 Product Liquid Sampling System Controls

Auditing material inventory and verifying UF_6 product quality are done by direct sampling of product material which has been liquefied in a specially designed sampling autoclave. Process control and protection systems used with this autoclave are represented in Figure 5.6, which simplifies the actual system and does not identify signal paths for all process control and protection circuits.

The primary process control of the sampling autoclave affects UF_6 liquefaction. The heating rate is determined by monitoring the cylinder exit line pressure, which is directly related to UF_6 temperature and phase state. The autoclave state switch has two states: heater on and heater off, and the operator manually sets the position of the cylinder valve. The temperature and pressure of main plant cooling water are monitored to control the flow and temperature of water used to cool and solidify the liquid UF_6 after sampling is complete.

The primary protection function of the product sampling systems is prevention of UF_6 release from the process equipment. Release mechanisms include breaking of the confinement barriers created by external force or internal expansion of the UF_6. Internal threat to the confinement barriers is indicated by elevated pressure or temperature, or by excess UF_6 inventory. Instrumentation and controls monitor variables which indicate threat to the confinement barriers, prevent occurrence of a potentially dangerous situation, or take mitigative action. Protective controls for the sampling autoclave are presented in Tables 5.11. In addition, an HF monitor on the GEVS vent line is used to check for leaks before the autoclave is opened. Set points for response of the product sampling autoclave protection systems are:

- Autoclave air temperature of 127 °C (260 °F)
- Autoclave air pressure of 170,300 pascals (24.7 psia)
- Cylinder pressure of 698,480 pascals (100 psia)
- Heater element temperature of 150 °C (302 °F).

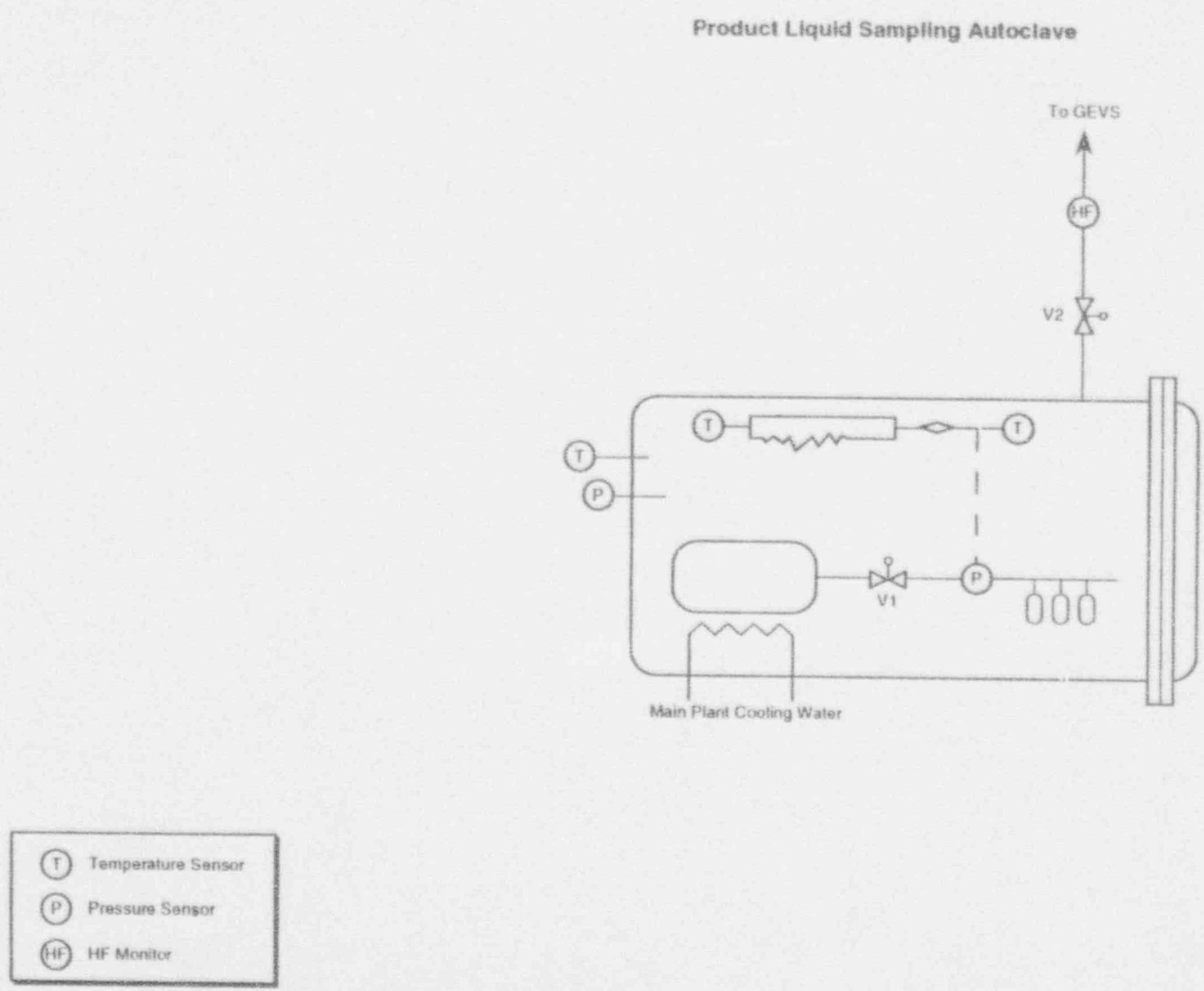

Figure 5.6 Schematic for product sampling autoclave instrumentation and controls

Table 5.11 Sampling autoclave protective controls

Function	Monitored Variable
De-energize heaters	• high autoclave pressure, redundant • high autoclave temperature, redundant • high heater element temperature • high cylinder exit line pressure • incorrect position of autoclave exit line valve
Prevent door opening	• position of GEVS vent line valve • high autoclave air pressure
Isolate from GEVS	• position of GEVS vent line valve

5.1.7 Product Blending System Controls

The blending system enables CEC to meet customer specifications. A single blending system serves the three CEC plant units. The major elements of the blending system are the autoclaves, desublimers, receiver cylinder stations, and the associated piping and valves. The primary process control circuits used in the blending and venting operations are represented schematically in Figure 5.7, which simplifies the actual system and does not identify all signal paths of the process control and protection circuits. Coordinated, upper-level control of the purification and feed operations is provided by state switches on the autoclave, desublimer, and receiver cylinder station control panels. Systems activated by the autoclave, desublimer, or receiver cylinder state switches are specific to the function performed in that state. The autoclave switch has five states, the desublimer switch has six states, and the receiver cylinder station switch has four states. The autoclave, desublimer, and receiver cylinder station states and representative valve positions are summarized in Tables 5.12, 5.13, and 5.14, respectively. The operator opens or closes the valves (their position generally remains unchanged during the particular operation), activates control systems and interlocks, and activates or inhibits the appropriate protection systems.

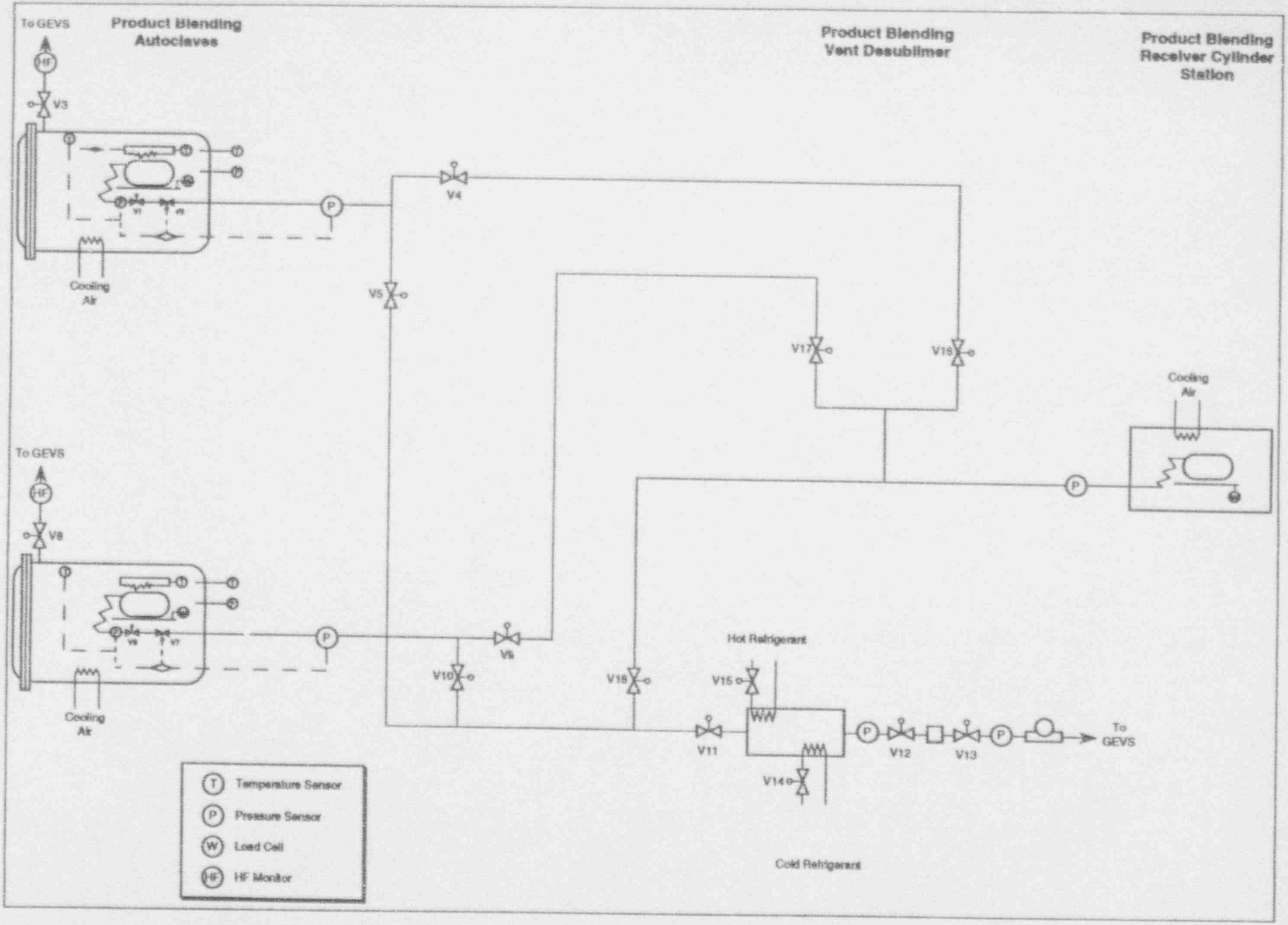

Figure 5.7 Schematic for product blending system instrumentation and controls

Table 5.12 Blending autoclave states and valve positions

State	Valve Position			
	V1/V6	V2/V7	V4/V9	V5/V10
Isolate	closed	closed	closed	closed
Vent	open	open	closed	open
Heat-up	closed	closed	closed	closed
Transfer	open	open	open	closed
Heels transfer	open	open	closed	open

Table 5.13 Blending vent desublimer states and valve positions

State	Valve Position				
	V11	V12	V13	V14	V15
Heat	closed	closed	closed	closed	open
Gas-over	open	closed	closed	closed	open
Chill	closed	closed	closed	open	closed
Standby	closed	open	**	**	closed
On-line	**	**	**	open	closed
Vent	open	closed	closed	open	closed

** controlled desublimer pressure

Table 5.14 Product blending receiver cylinder states and valve positions

State	Valve Position		
	V16	V17	V18
Transfer 1	open	closed	closed
Transfer 2	closed	open	closed
Vent	closed	closed	open
Isolate	closed	closed	closed

Table 5.15 Blending autoclave protective controls

Function	Monitored Variable
De-energize heaters	• high autoclave pressure, redundant • high autoclave temperature, redundant • high heater element temperature • high cylinder exit line pressure • incorrect position of autoclave exit line valve
Terminate flow to plant unit header	• high autoclave exit line pressure, redundant
Prevent door opening	• position of GEVS vent line valve • high autoclave pressure
Isolate from GEVS	• position of GEVS vent line valve
Alarm on excess cylinder weight	• autoclave load cell

Table 5.16 Blending desublimer vent protective controls

Function	Monitored Variable
Terminate inlet flow	• high desublimer pressure
Terminate outlet flow	• low desublimer pressure • high vacuum pump inlet pressure

Table 5.17 Blending receiver cylinder station protective controls

Function	Monitored Variable
Terminate inlet flow	• excess cylinder weight

controllers connected to each of these four sensors are independently capable of terminating the heater function. The control logic for the autoclave Class I protection instrumentation is presented in Figure 5.8. By virtue of the immediate relation of autoclave air space temperature and pressure to UF_6 temperature and pressure, the system is designed to sense potentially hazardous conditions. because it can stop heat inflow, the system is designed to ensure that acceptable operating limits are not exceeded. The limits of autoclave air space temperature and cylinder pressure are within the design limits of the cylinders and are therefore acceptable. With threefold redundancy, the protection systems are designed to be reliable by maintaining function despite the loss of any component. On loss of power, the heater coils, as non-essential loads stop producing heat and the protection systems, as essential loads, continue to function. On the basis of the above description and analysis, the NRC staff concludes that the Class I instrumentation of the feed, blending, and sampling autoclaves meet the requirements of the ANPR and are adequate to protect worker and public health and safety.

5.3 NRC Staff Evaluation of Controls for Process Shutdown

The NRC staff has reviewed the CEC design to identify the instrumentation, controls, and equipment required to achieve and maintain safe shutdown. The ANPR provides guidance on three aspects of plant operation relevant to safe shutdown. First, a control room should operate and, when required, shut down the plant and maintain the shut down plant in a safe condition. The capability to shut down the plant should not be affected by the loss of any one control area. Second, protection systems should sense potentially hazardous conditions and activate equipment required to ensure safety of operating personnel and the public. These protection systems should satisfy a single-failure criterion and fail to a safe state. Third, instrumentation and control systems should monitor variables and systems important to safety for safe shutdown conditions. The systems required for safe shutdown are those which terminate evaporation of the feed UF_6 and those which remove the process inventory of process equipment. The CEC design provides for safe shutdown of the plant from LCCs or from the CCR. The following sections evaluate, plant system by plant system, the shutdown capability of protection and important to safety systems.

The major step in shutdown of the plant is termination of transfer of UF_6 to the cascades. The equipment layout and controls used to shut off feed are represented in Figure 5.1. The heaters used to evaporate UF_6 may be tripped by using a hand switch or by setting the autoclave state switch in the isolate position. Selecting the isolate state also closes the two exit line valves inside the autoclave (V1 and V2), and the plant unit header and desublimer vent valves (V4 and V5). The potentially hazardous condition of elevated line pressure is monitored by redundant pressure sensors. Elevated line pressure closes valve V1 inside the autoclave. Important-to-safety systems in the feed system are the protection loops, which trip the heaters on signals of high temperature or pressure in the autoclave air space. These variables are monitored continuously, and the important-to-safety systems are fully redundant,

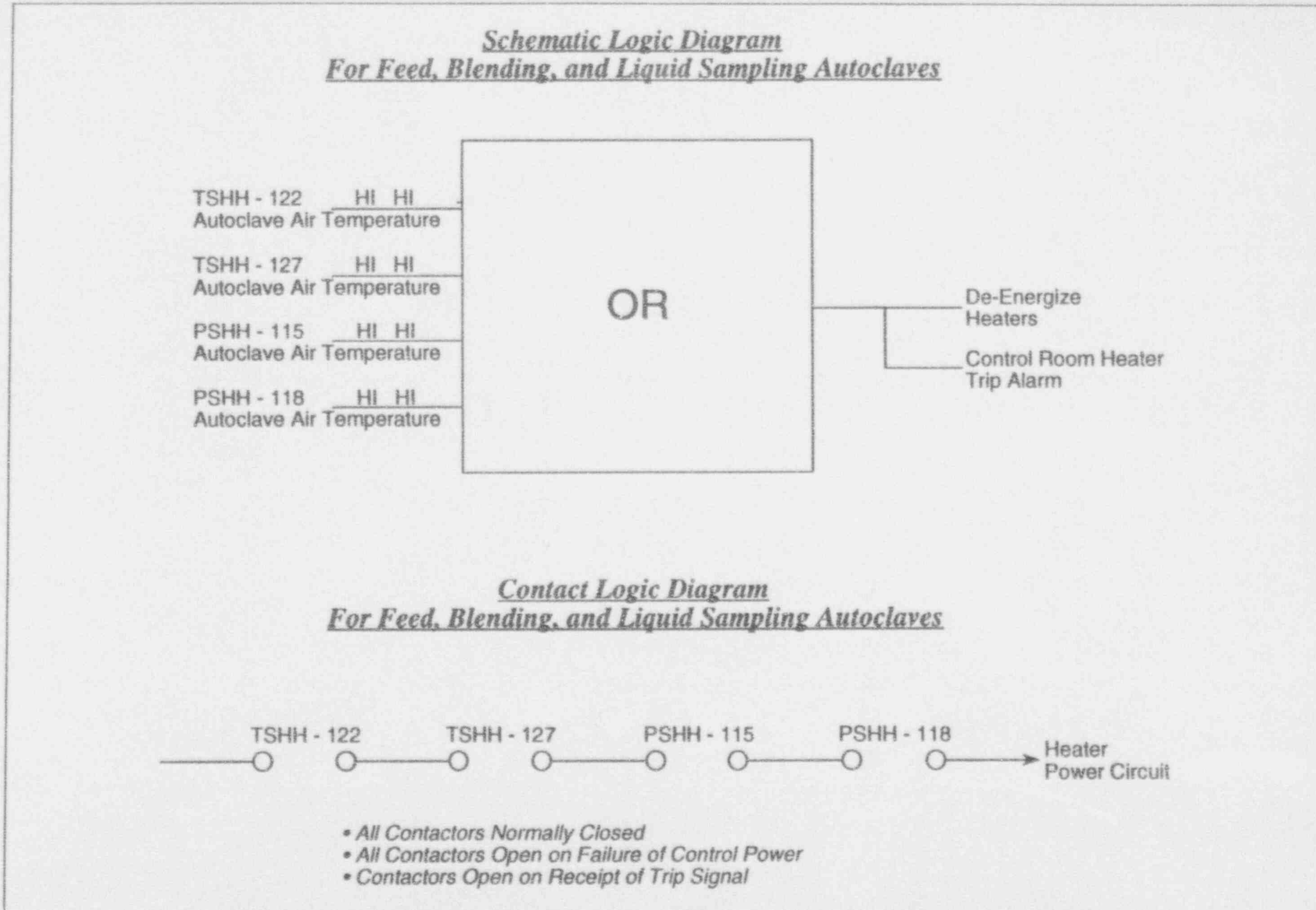

Figure 5.8 Logic diagrams for autoclave class I protection systems

and are displayed and alarmed locally and in the control room. Shutdown of the plant with UF_6 in the feed purification desublimer does not constitute a hazardous situation, but setting the desublimer state switch in the chill position closes the desublimer inlet and outlet valves (V6 and V7), and closes the hot refrigerant valve (V10). UF_6 may then be transferred to a cylinder in the purification cubicle at the direction of the operator. The potentially hazardous condition of elevated pressure in the desublimer is monitored, and the desublimer inlet and outlet valves are shut when pressure increases beyond a control point. The autoclave outlet line valves and the desublimer inlet and outlet valves fail to the closed position and thereby reduce the possibility of releasing hazardous material. For these reasons, the NRC staff concludes that provisions for shutdown of feed are in accord with ANPR provisions and are adequate for protection of worker and public health and safety.

Removing the cascade inventory can be done by dumping to the product and tails take-off stations or by means of the contingency dump system. In a dump to the product or tails cylinders, the low- and high-pressure vacuum pumps continue to operate. For the series arrangement of low-pressure pumps, a single pump can handle the full flow. For the series arrangement of high-pressure pumps, any two of the three pumps can handle the full flow. Thus, the failure of a single pump does not prevent dumping of the inventory. Pressure sensors in the lines detect the potentially hazardous condition of high pressure and isolate the affected system components. Temperature sensors on the high-pressure pumps indicate pump overload and alert the operator to trip the pump. The take-off systems have five product cylinder stations and five tails cylinder stations. Because of the low inventory of the cascades, the loss of a single component does not prevent dump of the inventory to either the product or tails take-off stations. Pressure sensors and load cells at the product and tails take-off stations sense potentially dangerous conditions and isolate the affected cylinder. The product vent desublimer can be placed in a safe condition by setting the state switch in the chill position. The potentially dangerous condition of elevated pressure in the desublimer is monitored, and the inlet and outlet valves are closed on excess pressure. If both the product and tails take-off stations are unavailable, the cascades can be dumped to the contingency dump system. For these reasons, the NRC staff concludes that the proposed systems for removing plant inventory for shutdown are in accord with ANPR guidance and are adequate for protection of worker and public health and safety.

Additional UF_6 may be present in the product blending and sampling systems at the time of a plant shutdown. Setting the blending autoclave state switches in the isolate position shuts off the heaters and closes the two autoclave exit line valves and the vent and transfer line valves (V1 and V6, V2 and V7, V4 and V9 and V5 and V10). Heaters may also be shut off by separate hand switches. The potentially dangerous condition of elevated blending autoclave exit line pressure is monitored, and the autoclave exit valve is closed on a signal indicating high pressure. Blending autoclave exit valves and desublimer inlet and outlet valves fail to the closed position, isolating the UF_6 inventory. Setting the sampling autoclave state switch in the off position shuts off the heaters. For the blending and sampling autoclaves, redundant important-to-safety systems continuously monitor and display autoclave air pressure and temperature. The heaters are de-energized on a signal indicating high pressure or temperature. For these reasons, the NRC staff concludes that the designs of the blending and sampling systems for shutdown are in accord with ANPR guidance and are adequate for protection of worker and public health and safety.

5.4 NRC Staff Evaluation of Controls for Process Operation

The CEC design includes process control and protection systems for safe operation under normal, off-normal, and accident conditions. The ANPR provides guidance on three aspects of the function of controls for process operation. First, the confinement system and barriers and other systems affecting overall plant safety should be monitored. Second, controls should be provided to maintain process variables within prescribed operating ranges under normal conditions. Third, the control systems should be designed to fail to a safe state. The process control systems described above for process shutdown are the same systems used to control normal operations. In general, each individual process controller has three set points. The lower two set points are used to control process operation while the third setpoint is used to shut the controlled element. State switches at LCCs have settings which define valve positions and interlocks for normal operating conditions in addition to the settings described above which are appropriate for shutdown conditions. In conjunction with these design features, the evaluations of individual systems presented above for process shutdown thus also support the findings that confinement barriers are monitored, that appropriate variables are maintained within specified ranges, and that systems fail to a safe state for normal, abnormal, and accident conditions. For these reasons, the NRC staff concludes that the proposed instrumentation and controls are in accord with the ANPR guidance for normal, off-normal, and accident conditions and are adequate for protection of worker and public health and safety.

5.5 NRC Staff Evaluation of Control Room Instrumentation

The CEC CCR is an information-processing and response center which can monitor and control all variables monitored and controlled by LCCs, and all monitored variables alarmed at the local level are alarmed in the CCR. A redundant data highway transmits information from local instrumentation to the CCR, which can display and store the data. Consoles display the data in process-mimic form; keyboards are used to change controlled equipment. Optical disk equipment provides redundant permanent data storage. Two-way radios are available for voice communication with LCCs when necessary.

A control room must permit occupancy under normal, off-normal, and accident conditions and provide safe plant operation or shut down. The CCR is physically separated from potentially contaminated areas of the plant and is served by a ventilation system independent of the ventilation systems serving potentially contaminated areas. An alternative system should provide for safe shutdown if the control room is not available. Given the CCR design and the considerations of SER Section 5.3 on safe shutdown, the NRC staff concludes that the CEC CCR design is in accord with ANPR guidance and is adequate for protection of worker and public health and safety.

6 AUXILIARY SYSTEMS

Claiborne Enrichment Center (CEC) auxiliary systems include equipment the function of which is not required for confinement of uranium hexafluoride (UF_6) process material but which support operation of separation equipment and over-all operation of the facility. These auxiliary functions include maintenance of building environmental conditions, provision of electrical power, supply and removal of process heat, and supply of instrument air and nitrogen purge gas. This chapter describes these four classes of auxiliary systems and presents a finding on adequacy relative to regulatory requirements of Part 70 of Title 10 of the Code of Federal Regulations (CFR), the guidance of the Advance Notice of Proposed Rulemaking for 10 CFR Part 76 (ANPR) (NRC, 1988a), or the requirements of standard industrial practice. Additional support systems involved in control of release of radioactive material such as waste management systems and systems with potential safety significance are described in Chapters 3, 4, and 7 of this Safety Evaluation Report (SER). Process descriptions presented in this chapter are drawn from the CEC Safety Analysis Report (SAR) (LES, 1993a) and Environmental Report (ER) (LES, 1993b). Analyses and evaluations presented in this chapter are independent NRC staff analyses and evaluations.

6.1 Ventilation Systems

The CEC Separations Building is separated by design into three plant units and a set of support areas. For the purposes of general heating, ventilation, and air conditioning (HVAC), each plant unit is divided into three areas which are nominally identical to analogous areas of the other plant units. The three process areas for each plant unit are UF_6 Handling and Auxiliary Areas, Cascade Halls, and Electrical Distribution Area. The four support areas are Technical Services Area, Blending Facility and Cylinder Handling Area, Utility Area, and Control and Administration Area.

Each of the thirteen areas listed above has an independent ventilation system which provides general purpose heating and cooling for that area. In addition, the three plant units are served by a common process gas clean-up system, the Gaseous Effluent Vent System (GEVS). Of these fourteen systems, only the GEVS and a portion of the TSA HVAC system handle potentially contaminated air. The GEVS and TSA HVAC system are described and reviewed in Section 4.9 of this SER. This section describes HVAC systems serving six types of areas: the three areas replicated in each plant unit and three additional support areas. Each of these areas uses the flow configuration shown in Figure 6.1.

<u>UF_6 Handling and Auxiliary Area</u>

The UF_6 Handling Area houses the UF_6 feed and purification systems, the product and tails take-off systems, and the product sampling systems. The auxiliary area houses the process heating and cooling systems which support the feed and take-off equipment. Fresh outside air

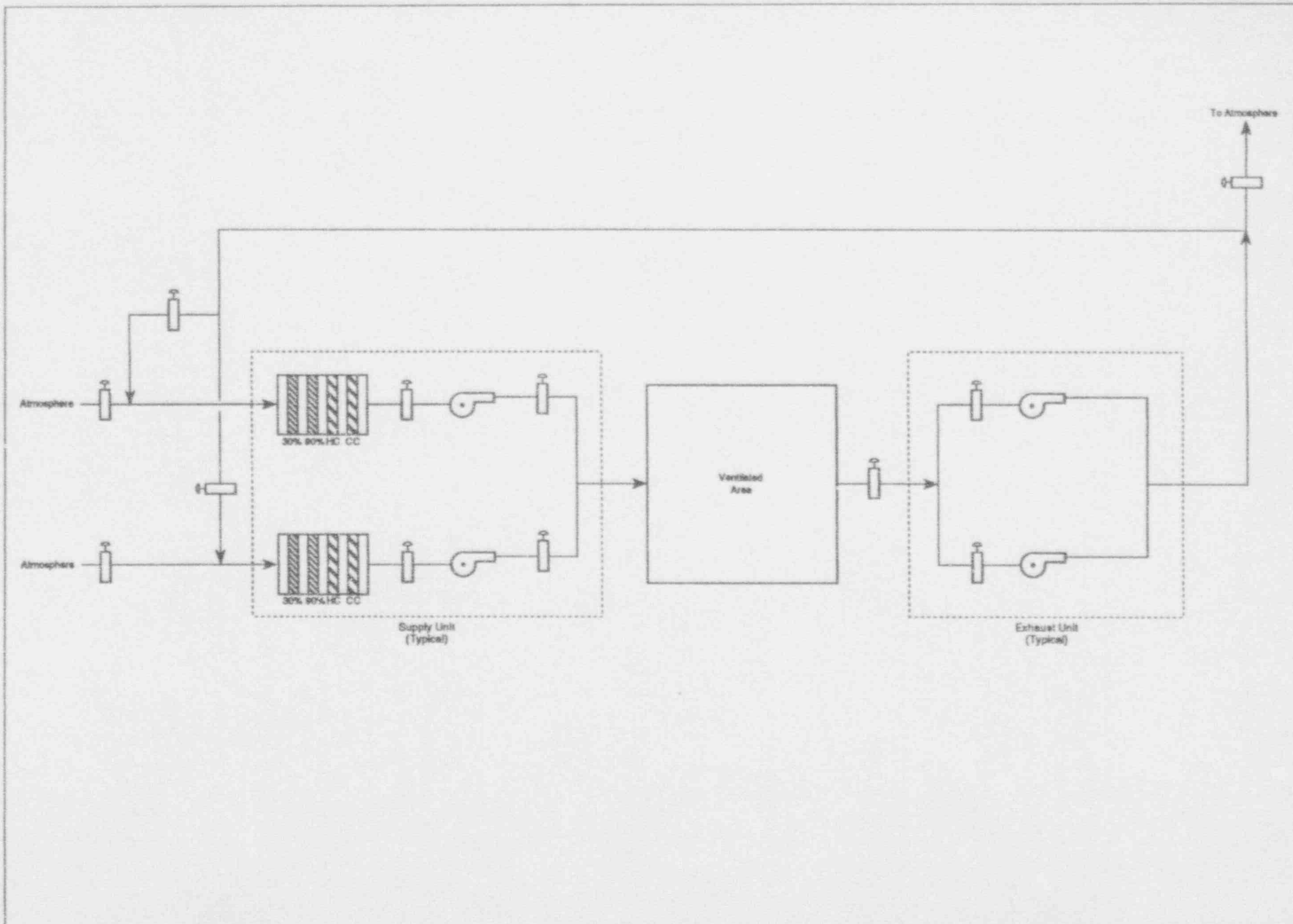

Figure 6.1 CEC ventilation systems flow schematic

is drawn into the system through the supply unit comprised of three parallel paths each containing two filters, heating and cooling coils, and a fan. Each air conditioning unit and fan handle one-third of the total flow of 35 m^3/s (75,600 cfm). The combined flow is distributed through the ventilated process areas to three parallel exhaust fans. Ten percent of the exhaust fan discharge is released to the atmosphere while the balance of the flow is returned to the supply fans. The volume of the ventilated area is approximately 28,800 m^3 (1.02 million ft^3) yielding a change-over rate of 4.4 volumes per hour. The system maintains temperature in the UF_6 Handling Area between 19 and 22 °C (66 and 72 °F) and in the Auxiliary Area between 15 and 50 °C (59 and 122 °F).

Cascade Halls

The Cascade Hall ventilation system for each plant unit is comprised of two identical sub-systems which each service the seven cascades of one assay unit. For each sub-system, fresh air is drawn into a supply unit comprised of two parallel air conditioning units. Each air conditioning unit contains two filters, heating and cooling coils, and a fan. Total supply flow is 18.5 m^3/s (39,250 cfm). Supply air is routed through the cascade halls and pulled through two parallel exhaust fans. Ten percent of the exhaust fan flow is discharged to the atmosphere while the balance is returned to the supply inlet. Air temperature is maintained between 19 and 30 °C (66 and 85 °F) and relative humidity is maintained below seventy percent. Total volume of the seven cascade halls is 36,440 m^3 (1.29 million ft^3) which, in combination with the cited supply flow, yields a change-over rate of 1.8 volumes per hour.

Electrical Distribution Area

Each plant unit has an electrical distribution area which supplies power to the equipment and cascades of that plant unit. Fresh air is drawn into the area HVAC system through a single air handling unit by two parallel supply fans. The air handling unit is comprised of two filters, and heating and cooling coils while each of the two supply fans handles half of the supply flow of 3.9 m^3/s (8300 cfm). The supply flow is routed through the electrical distribution area and the access corridor to two parallel exhaust fans. Ten percent of the exhaust flow is released to the atmosphere while the balance is returned to the supply unit. Air temperature is maintained between 19 and 30 °C (66 and 86 °F) with a change-over rate of 1.8 volumes per hour.

Blending Facility and Cylinder Handling Area

Feed cylinders are transferred into the Separations Building through the Cylinder Handling Area and product cylinders are filled to a specified assay from donor cylinders in the Blending Area. Ventilation air for these areas is drawn into two parallel supply units each comprised of two filters, cooling and heating coils, and a fan. Each fan handles half of the supply flow of 15.5 m^3/s (32,780 cfm). Supply air is passed through the process areas to two parallel exhaust fans. Ten percent of the exhaust air is released to the atmosphere while the balance is returned to the supply units. Air temperature is maintained between 19 and 30 °C (66 and 86 °F) with a change-over rate of 3.1 volumes per hour.

Utility Area

The Utility Area of the Separations Building contains the HVAC equipment rooms and the mass spectrometer room. Fresh air for these areas is drawn into a single air handling unit by two parallel supply fans. The air conditioning unit is comprised of two filters and heating and cooling coils. Each of the supply fans is sized to handle half of the total flow of 9.8 m^3/s (20,660 cfm). Supply air is pulled through the utility areas by two parallel exhaust fans. Ten percent of the exhaust air is discharged to the atmosphere while the balance is returned to the supply unit. Air temperature is maintained between 8 and 49 °C (46 and 120 °F), with a change-over rate of 4.6 volumes per hour is achieved.

Control and Administration Area

Offices, conference rooms, the main control room, and an electrical room are ventilated by the Control and Administration Area HVAC system. Fresh air is drawn into a single air handling unit by two parallel supply fans. Each fan handles half of the supply flow of 7.9 m^3/s (14,750 cfm). Supply air is distributed to the rooms and returned to two parallel exhaust fans. Ten percent of the exhaust flow is discharged to the atmosphere while the balance is returned to the supply unit. Air temperature in conference and control rooms is maintained between 21 and 24 °C (70 and 75 °F) and relative humidity is maintained between forty and fifty percent. Temperature in an electrical room ventilated by this system is maintained between 7 and 49 °C (45 and 120 °F), with an average air change-over rate of 2.6 volumes per hour.

NRC Staff HVAC Systems Evaluation

Ventilation systems described in this section are not designed to fulfill a safety function and are shut down in the event of a UF_6 release in the respective area. All components and duct work are constructed of galvanized steel in accord with industrial (Sheet Metal and Air Conditioning Contractors National Association) standards. Given the design description summarized above and the non-safety function of the systems, the NRC staff finds that the design of the systems meets the ANPR requirement to maintain adequate environmental conditions and is consistent with safe operation of the CEC.

6.2 Emergency Power

Electricity is the sole source of energy used to drive normal operational, support, and safety systems at the CEC. Electrical power is delivered to the CEC on two independent, redundant overhead 115 kilovolt lines from the Louisiana Power and Light (LP&L) grid system. The first line is from a LP&L substation located in Haynesville, LA, 20 kilometers (12 miles) northwest of the CEC while the second line is from a LP&L substation located at Bernice, LA, 30 kilometers (19 miles) east of the CEC. Incoming voltage is reduced from 115 to 13.8 kilovolts through two main transformers. The main transformers feed primary and reserve 13.8 kilovolt switchgear buses located in the Separations Building TSA. Power from the 13.8 kilovolt source lines is reduced to the 480 volt level in four ring circuits serving 21 transformers. Nine transformers fed from ring circuits 1 and 2 distribute power through a motor control center to non-essential drive systems located in the Electrical Distribution Room. Four transformers fed from ring circuit 3 and 4 support non-essential loads while eight transformers of ring 3 and 4 circuit support essential loads. The essential loads are :

- Product and tails high-pressure vacuum pumps
- Product and tails low-pressure vacuum pumps
- Pipe electric heat tracing
- Hot box heaters
- Uninterruptable power supply
- Air fan motor for Air Cooling Product Cylinders Subsystem
- Air fan motor in GEVS
- Diesel fuel tank pump motor for Standby Generator System
- Circulation pump motors in Hot and Cold Refrigerant Systems
- Communications System
- Outlets for mobile pump sets.

Essential equipment is fed from motor control centers which have back-up power from the Standby Diesel Generator System located in a dedicated building adjacent to the Separations Building. The Standby Generator System is comprised of two package generator units, two aboveground 37,850-liter (10,000-gallon) storage tanks, and associated controls. Each of the two 100 percent load generators provides 1,500 kilowatts of 480 volt power within 20 seconds of primary power interruption. Fuel storage capacity is adequate for 7 days operation at rated load. The generator starts automatically if power is lost to an essential load or if power is lost to the control panels of both diesel generators. The generator is shut down manually when normal operating power is available to all essential loads. The generator system is operated periodically to assure availability in loss of off-site power.

In addition to standby diesel generator power for essential loads, the design of the CEC includes uninterruptable battery power for critical systems. The critical systems are computer system, instrument power supply, Contingency Dump System vacuum pumps, and emergency lighting. Each of these systems is normally fed from an essential load switchgear but in the event of failure of the standby power system battery power is supplied from an Uninterruptable Power System (UPS) bus. The UPS is designed to supply power for a period of approximately 15 minutes to allow shutdown in the event of total loss of power. Back-up batteries are located in the electrical distribution room of each cascade and in the electrical room of the administrative area. The battery rooms are ventilated to remove the risk of explosion because of the build-up of hydrogen gas. Failure of ventilation fans is alarmed in the Central Control Room.

NRC Staff Evaluation of Emergency Power Systems

Part 70 of Title 10 of the Code of Federal Regulations specifies that license applications contain description of equipment and facilities used to protect health and minimize danger to life and property. The ANPR specifies that emergency power shall be provided to instruments, confinement systems, utility service systems, and process systems to allow safe shutdown of the facility. Based on the description of the emergency power system summarized in this section and on review of the design of this system the NRC staff finds that the system meets the requirements of 10 CFR Part 70 and the ANPR, and is adequate for protection of public health and safety.

6.3 Process Heat Exchange

Energy is input to CEC Separations Building systems in the form of electrically generated heat, as process heat from the hot refrigerant system, and as warm make-up ventilation air. Energy from each of these sources is indirectly removed using Main Plant Cooling Water (MPCW) Systems servicing each plant unit. This section describes the MPCW Systems and the individual sub-systems serviced by the MPCW systems.

Main Plant Cooling Water Systems

MPCW Systems are closed-circuit cooling systems which discharge process heat to the atmosphere through air-cooled chillers using Freon (R-22) as the heat transfer medium. The systems for each of the three plant units are nominally identical except that the MPCW System for Plant Unit 1 services the Blending Area and the Auxiliary Areas. System loads on the Plant Unit 1 MPCW system are Machine Cooling Water (MCW) Systems, HVAC Systems, Spray Cooling Water Systems, Cold Refrigerant System, Product Blending System, Product Liquid Sampling System, and Plant Instrument Air System.

Plant Units 2 and 3 service similar equipment except for the Blending Area systems, the Cold Refrigerant System, and the Instrument Air System. Each MPCW system is sized to supply cooling water at 8 °C (46 °F) and is capable of transferring 1,460 kilowatts ($5.0\times10^{+6}$ BTU/hr). The MPCW systems are located in the Auxiliary Areas and are comprised of an expansion water tank, two water chillers, two pumps, and associated piping. Under normal operating conditions cooling water supply temperature is monitored and is maintained at 8 °C (46 °F) and chillers are shifted on- and off-line to meet the transient load. When offsite power or instruments fail, the system shuts down and is not required to maintain the plant in a safe condition.

Machine Cooling Water System

An MCW System is provided for each assay unit of each plant unit. The function of the closed loop MCW system is to transfer heat from the centrifuges to the MPCW systems for discharge to the atmosphere. The MCW system for a plant unit circulates demineralized water using a MPCW heat exchanger and an expansion tank serving both assay units and pumps serving individual cascades. The MCW systems are located in the Auxiliary Areas and Cascade Halls and are sized to transfer approximately 1,200 kilowatts ($4.1x10^{+6}$ BTU/hr) of energy to the MPCW system. Cooled, demineralized water flows from the MPCW heat exchanger to the cascades at a rate of 3,975 lpm (1,050 gpm). Water temperature entering the centrifuge cooling jackets is monitored and controlled by throttling flow. The MCW systems are not required to maintain the plant in a safe condition and are shutdown on loss of power.

Spray Cooling Water System

Each plant unit is provided with a Spray Cooling Water System (SCWS) to remove heat of desublimation in product, tails, blending, and purification cylinders. Direct contact with hot feed and tails cylinders is employed, but indirect contact with recirculated air is used to remove heat from product and blending cylinders. The SCWS comprises of an expansion tank, recirculation pumps, and piping to the load systems. Each plant unit SCWS is sized to circulate 1,040 lpm (275 gpm) of water and transfer 70 kilowatts (240,000 BTU/hr) of energy. The supply temperature of the spray water is 4 °C (39 °F) and the temperature of the cooled air is 10 °C (50 °F). Except for the expansion tank, all equipment in the SCWS is 100-percent spared. During a power failure, one water pump and one air fan continue to operate for two hours to desublime cascade inventory into tails or product cylinders.

Cold Refrigerant System

A Cold Refrigerant System located in the Auxiliary Areas is provided for each plant unit. The system services the desublimers in the feed purification, product take-off, and product blending vent systems. The cold refrigerant system comprises of a refrigerant chiller, a circulation pump, an expansion vessel, and associated piping and controls. The chiller uses a second Freon heat transfer fluid to reject heat to the MPCW system at a maximum rate of 12.7 kilowatts (43,200 BTU/hr). A flow of 208 lpm (55 gpm) is provided at a temperature of -70 °C (-94 °F). Refrigerant supply temperature is monitored and controller by varying the chiller load capacity. Pressure differential between the cold refrigerant supply and return lines is monitored, and constant flow rate is maintained by varying by-pass flow. Pressure and level in the expansion vessel are monitored and alarmed in the control room. A spare chiller is provided to assure system reliability. Loss of main plant cooling water initiates system shutdown except that the recirculation pumps continue to run to remove residual heat. Interconnection of desublimers for the plant units provides redundancy to protect against loss of any one Cold Refrigerant System. Loss of the system does not pose a safety concern due to back-up capability and automatic shutdown of supply function.

Hot Refrigerant System

A Hot Refrigerant System supplied for each plant unit is used to heat desublimers in the feed purification, product take-off, and product blending vent systems. The vapor UF_6 is transferred to feed or product cylinders where it is solidified. The Hot Refrigerant System is comprised of a refrigerant heater, recirculation pump, expansion vessel, and associated piping. Freon (R-11) refrigerant is circulated at a rate of 260 lpm (69 gpm) at a supply temperature of 50 °C (122 °F). Energy is supplied to the heater with hot water and the heater is rated at 2,930 kilowatts ($1.0x10^{+7}$ BTU/hr). Refrigerant supply temperature is monitored and controlled through variation of heater hot water supply. Expansion vessel pressure is monitored and controlled by addition of heat from external heat pads. Failure of the systems results in the loss of the ability to empty desublimers but does not pose a safety hazard.

6.4 Process Gas

CEC instruments and controls are operated by using pressurized air and various process lines are purged periodically using nitrogen gas. This section describes the design and operation of these systems.

Instrument Air

A single Plant/Instrument Air System provides 0.8 MPa (100 psig) compressed air to instruments, controls, and equipment of the CEC. The system comprises of two moisture separators, two package series compressor units, two dryer units, and two air receiver vessels. Each of the moisture separators, compressors, and dryers is sized to handle 100 percent of the design flow of 5.4 standard m^3/min (190 scfm). Atmospheric air is pre-cooled, compressed, dryed, and stored for delivery as needed to system loads. Under normal conditions the two compressors operate at reduced load and the spare moisture separators and dryers are off-line. The receiver vessels are each sized to provide 10 minutes' supply of average plant requirement with no flow from the compressors. Moisture, temperature, and pressure alarms in the control room indicate off-normal or failure conditions. On total loss of the system, the system hold-up capacity is sufficient to allow safe shutdown of the facility.

Nitrogen Gas

Liquid and gaseous nitrogen are used in the separation process to collect samples, to blanket equipment, and to purge process lines. A single nitrogen supply system located adjacent to the Separations Building is provided to meet these requirements. The system is comprised of a 37,850-liter (10,000-gallon) cryogenic storage tank, nitrogen vaporizer, heater, and associated piping and controls. Liquid nitrogen is delivered to the storage tank by truck. An insulated pipe transfers liquid nitrogen from the storage tank to a dispensing station located in the Separations Building. A second pipe transfers gaseous nitrogen from the vaporizer and heater to process users. The system is capable of providing 30 days supply to process users at a maximum gas flow rate of 2.8 standard m^3/min (100 scfm). Storage tank pressure is monitored, and the tank is vented to the atmosphere. Operation of the system is not important to safety and is not required for safe shutdown of the facility.

7 RADIOACTIVE AND CHEMICAL WASTE MANAGEMENT

Operating the Claiborne Enrichment Center (CEC) will produce gaseous, liquid, and solid waste effluent streams. Some of these streams will and others will not contain hazardous or radioactive chemical components, either alone or in combination. This section describes the sources, characteristics, and quantities of potentially harmful materials which the proposed CEC could release to the environment, estimates the resulting concentrations in environmental media, and compares these concentrations to limits specified in the Code of Federal regulations (CFR). The applicable limits are expressed as concentrations in 10 CFR Part 20, Appendix B, Table 2 (NRC, 1991c). This section also describes and evaluates the effluent monitoring program, and reviews the conformance of facility operation with the As Low As is Reasonably Achievable (ALARA) principle. The bases for evaluation of the effluent monitoring systems are the 10 CFR 20.1501 general requirements and the Advance Notice of Proposed Rulemaking for 10 CFR Part 76 (ANPR) (NRC, 1988a) requirement that radiological alarm systems be provided to indicate concentrations above control limits. The process descriptions presented in this chapter are drawn from the CEC Safety Analysis Report (SAR) (LES, 1993a) and Environmental Report (ER) (LES, 1993b). The analyses and evaluations presented in this chapter are the NRC staff's independent analyses and evaluations.

7.1 Gaseous Waste Management Systems

Gaseous streams released to the environment during normal CEC operation include the exhausts from general purpose heating, ventilation, and air conditioning (HVAC) systems serving each of the buildings and exhaust from the Gaseous Effluent Vent System (GEVS), or process off-gas system, serving equipment of the Separations Building. HVAC systems serving the Separations Building and the Centrifuge Assembly Building (CAB) may release minor amounts of chemically hazardous material to the environment. The GEVS and the HVAC system serving the Technical Services Area (TSA) of the Separations Building will probably release radioactive components to the environment.

Review criteria applicable to ventilation and off-gas systems include design criteria specified in the ANPR and release concentrations specified in 10 CFR Part 20. SER Section 8.5 considers conformance with regulatory requirements for dose.

7.1.1 Separations Building Gaseous Effluents

Thirteen subsystems serving independent areas will provide general ventilation for the Separations Building. All spaces are considered radiologically clean except for the portion of the TSA which is potentially contaminated. SER Sections 4.10 and 6.1 describe these systems in detail; this section describes only the principal relevant characteristics. Each of the thirteen subsystems has an air-conditioning unit which comprises a pre-filter, heating and cooling coils, and fans arranged in parallel. For clean areas, the present design calls for a total conditioned flow of 10 percent fresh air and 90 percent recycled air. Air exhausted from the clean areas is released to the plant stack without additional filtration. For the portion of the TSA which is likely to be contaminated, ventilation uses once-through flow, and the exhaust air passes through a pre-filter and HEPA filter before it is released to the plant stack. Total exhaust air flow rate at the plant stack is approximately 39 m^3/s, and changeover ratios for individual subsystems range from 1.8 to 4.6 volumes per hour.

The ANPR stipulates that air flow direction shall be maintained under normal and accident conditions, that variation in operating conditions be accommodated, and that normal and accident condition off-gases be safely controlled. Variation in operating conditions is accommodated through basic system design. The CEC design approach to maintaining operation and safe control of process off-gas uses monitoring of system function and airborne radioactivity to shut-down only those systems in affected areas and to maintain function in unaffected areas.

The GEVS collects gaseous radioactive material released from process equipment. Evacuation of piping connecting cylinders is the major source of radioactive material entering the system; minor contributions come from the feed, blending, and product vent systems. The GEVS is described in detail in SER Section 4.9; it includes a pre-filter, HEPA filter, and carbon adsorption bed for removing radioactive material. The GEVS releases an average flow of 1.84 m^3/s to the plant stack.

A conservative estimate of the average source term for CEC release of radioactive material to the atmosphere is 4.4 million becquerels (Bq) per year (120 microcuries per year). SER Section 7.4 discusses this estimate. The maximum annual average concentrations per unit source term (χ/Q) of 4.9×10^{-7} and 5.5×10^{-7} s/m^3 were determined for onsite and offsite receptors, respectively. These estimates were developed by using the XOQDOQ code described in SER Section 2.3. Combining the source term and χ/Q estimates produces an estimate of atmospheric uranium isotope concentrations of approximately 6.8×10^{-8} and 7.7×10^{-8} Bq/m^3 (1.9×10^{-18} and 2.0×10^{-18} microcuries/ml) for the maximum onsite and offsite receptors, respectively. These values are very small fractions of the 10 CFR Part 20 regulatory limits for release to unrestricted areas.

CEC operation routinely releases small quantities of hydrogen fluoride (HF) and trichlorotrifluoroethane (Freon 113) from the Separations Building stack. HF will be produced by hydrolysis of the uranium hexafluoride (UF_6) released to the desublimer vent and GEVS adsorption beds. The annual average HF release rate is estimated to be 6.35 kg/yr. Combining this release estimate with the maximum χ/Q yields an estimate of ambient HF concentration of 1.1×10^{-7} mg/m^3. Regulatory limits for HF air concentration have not been promulgated, but the estimated concentration is a factor of a million less than the American Conference of Governmental Industrial Hygienists (ACGIH) time-weighted average (TWA) threshold limit value of 2.5 mg/m^3 for occupational exposure to HF (ACGIH, 1986). The annual average release rate for Freon 113 is estimated to be 8640 kg/yr. Combining this release estimate with the χ/Q yields an estimate of 1.5×10^{-4} mg/m^3 for the ambient concentration of Freon 113. This value is also a factor of a million less than the ACGIH TWA threshold limit value of 7600 mg/m^3 for occupational exposure.

7.1.2 Centrifuge Assembly Building Gaseous Effluents

The CAB will be used to assemble, inspect, and test centrifuges before installing them in the Separations Building. A general HVAC system with no effluent filtration or monitoring is provided. Operations conducted in the CAB do not involve radioactive materials but do use and release acetone and Freon 113. Estimated release rates for acetone and Freon 113 are 100 and 400 kg/yr, respectively. The annual average value of χ/Q at the site boundary for ground-level release is estimated to be 3.1×10^{-4} s/m^3. Combining the release rates and the χ/Q values produced ambient air concentration estimates of 1.0×10^{-3} and 4.0×10^{-3} mg/m^3 for acetone and Freon 113, respectively. These concentrations are small fractions of the ACGIH TWA threshold limit values for these substances. Ambient concentrations resulting from the combined releases from the Separations Building and the CAB are less than the combined maximum concentrations predicted for these sources, 4.1×10^{-3} mg/m^3. This concentration is a small fraction of the ACGIH limit, and thus no adverse health impacts are expected.

7.1.3 NRC Staff Evaluation of Gaseous Effluent Waste Management

On the basis of this analysis, the NRC staff finds that radiological and nonradiological atmospheric releases related to CEC operation are consistent with the ANPR, the concentration guidelines of 10 CFR Part 20, and accepted health standards, and thus do not pose an undue risk to public health and safety.

7.2 Liquid Waste Management Systems

Liquid waste streams which are produced by routine CEC operation include aqueous liquids contaminated with uranium from Separations Building systems, noncontaminated sanitary and drain liquids from each CEC building, and small quantities of hazardous and mixed waste solutions. This section describes the liquid waste and liquid waste management systems and evaluates effluent releases against the 10 CFR Part 20 requirements.

7.2.1 Management of Aqueous Radiologically Contaminated Wastes

Aqueous streams contaminated with uranium in the Separations Building will flow from floor drains in the UF_6 Handling and Auxiliary Areas of each plant unit, TSA floor drains, the contaminated laundry system, and the equipment decontamination system. Estimates of the quantities and contamination levels for these streams presented in Table 7.1 indicate that the relatively small flow from the decontamination system will contain most of the uranium and that most of the streams will be contaminated to low levels.

Table 7.1 Radiologically contaminated liquid waste

Source	Volume (l/yr)	U Content (kg/yr)	U Concentration (mg/l)
Laundry Drain	681,300	0.11	0.15
Floor Drains	174,100	0.01	0.05
LWDS Dryer Flush Water	94,600	0.23	0.24
Laboratory Drains	70,000	0.01	0.13
Decontamination Rinse Water	12,500	0.46	70.0
Decontamination & Laboratory Solutions	10,200	80.0	3600.0

A block diagram of the Liquid Waste Disposal System (LWDS) proposed for treatment of the CEC radiologically contaminated aqueous waste is shown in Figure 7.1. The system operates on a batch-processing basis by using holding tanks and intermittent operations to produce a decontaminated liquid stream and a concentrated solid low-level waste stream. The initial step in the process precipitates uranium compounds from the citric acid decontamination solutions. Sodium hydroxide (NaOH) is added to increase the alkalinity and cause precipitation. The solids are separated from the resulting mixture by centrifugation, and the liquid is transferred to a surge tank. The clarified liquid from the precipitation process is mixed with the remaining contaminated liquid streams, filtered, and charged to a wiped-film evaporator. The decontaminated overhead vapor is condensed and transferred to a set of monitoring tanks. The concentrated evaporator bottoms are disposed of as low-level solid waste. If the uranium concentration in the condensate is less than 5 percent of the 10 CFR Part 20 limit for release to unrestricted areas, the batch is transferred to the sewage treatment system. Annual average flow through the Sewage Treatment System is ten times the flow of liquids processed in the LWDS. If the uranium content in the LWDS monitor tank liquid is greater than 5 percent of the 10 CFR Part 20 limit for release to unrestricted areas, the batch is passed through a mixed bed demineralizer and returned to the monitoring tanks. Liquids which do not meet the release limits are recycled to the evaporator feed tank and reprocessed.

7.2.2 Management of Industrial and Sanitary Wastewater

Slightly polluted industrial wastewater, sanitary sewage, and the monitored effluent of the LWDS will be processed in the CEC Sewage Treatment System. The annual average flow rate through this system is 9.5 million liters. A schematic representation of the system is

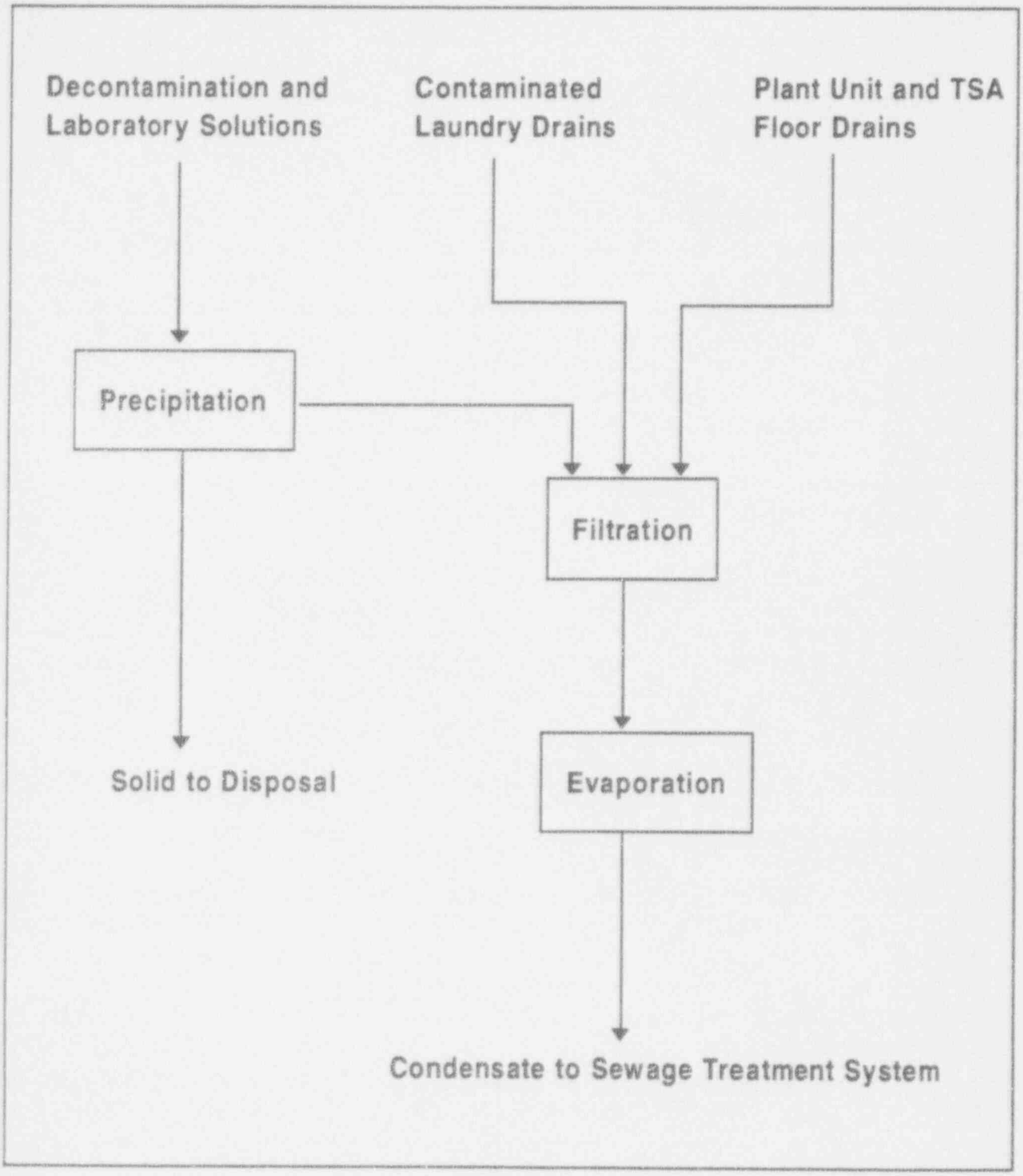

Figure 7.1 Schematic of the liquid waste disposal system

presented in Figure 7.2. Major means of processing include aeration, sludge digestion, filtration, chlorination, and settling. Effluents from the system include the treated wastewater and digested sludge which might contain small amounts of uranium compounds. The treated liquid is released to Bluegill Pond, and the sludge is disposed of in an offsite facility.

7.2.3 Management of Nonaqueous Liquid Radiological, Mixed, and Hazardous Waste

Radiolo ... nradiologically contaminated nonaqueous liquids are produced in small quantitie ... operation. These liquids include lubrication oils, solvents, laboratory chemic ... cellaneous materials. Miscellaneous materials include heavy oils and heat transf ... ne glycol, and Freon. The expected annual volume of radiologically contamin ... ardous liquid is 100 liters of hydrocarbon oil. Estimated annual volumes of liquid mixed waste are 25 liters of acetone and 600 liters of laboratory chemicals. Both classes of these liquids are contaminated at trace levels by uranium. Estimated annual volumes of nonradiologically contaminated liquids include 350 liters of oils, 80 liters of Freon 113, 15 liters of methanol, and 10 liters of perchloroethylene. Nonaqueous liquid wastes are collected at the point of generation and transferred to the waste storage area section of the TSA. Properly packaged and labeled shipments of these materials are transported offsite for treatment and disposal at authorized low-level radioactive, mixed, or hazardous waste treatment, storage, and disposal facilities.

7.2.4 NRC Staff Evaluation of Liquid Effluent Waste Management

On the basis of this analysis, the NRC staff finds that radiological and nonradiological liquid releases from CEC operation are consistent with the ANPR, the concentration guidelines of 10 CFR Part 20, and accepted health standards, and do not pose an undue risk to public health and safety.

7.3 Solid Waste Management Systems

7.3.1 Methods of Waste Management

Classes of solid waste which CEC operations produce include radioactive, mixed, hazardous, and industrial solid wastes. Each of these classes may be further categorized as wet or dry. Radiologically contaminated solid waste is Class A, as defined in 10 CFR Part 61 (NRC, 1982a). However, disposal of large quantities of depleted uranium tails was not contemplated in the development of 10 CFR Part 61 and near-surface disposal of this material is not considered appropriate. Identification, classification, and estimated generation rates of radioactive solid wastes are provided in Table 7.2. The solid waste disposal system is a set of handling procedures designed to collect, segregate, and dispose of the identified waste.

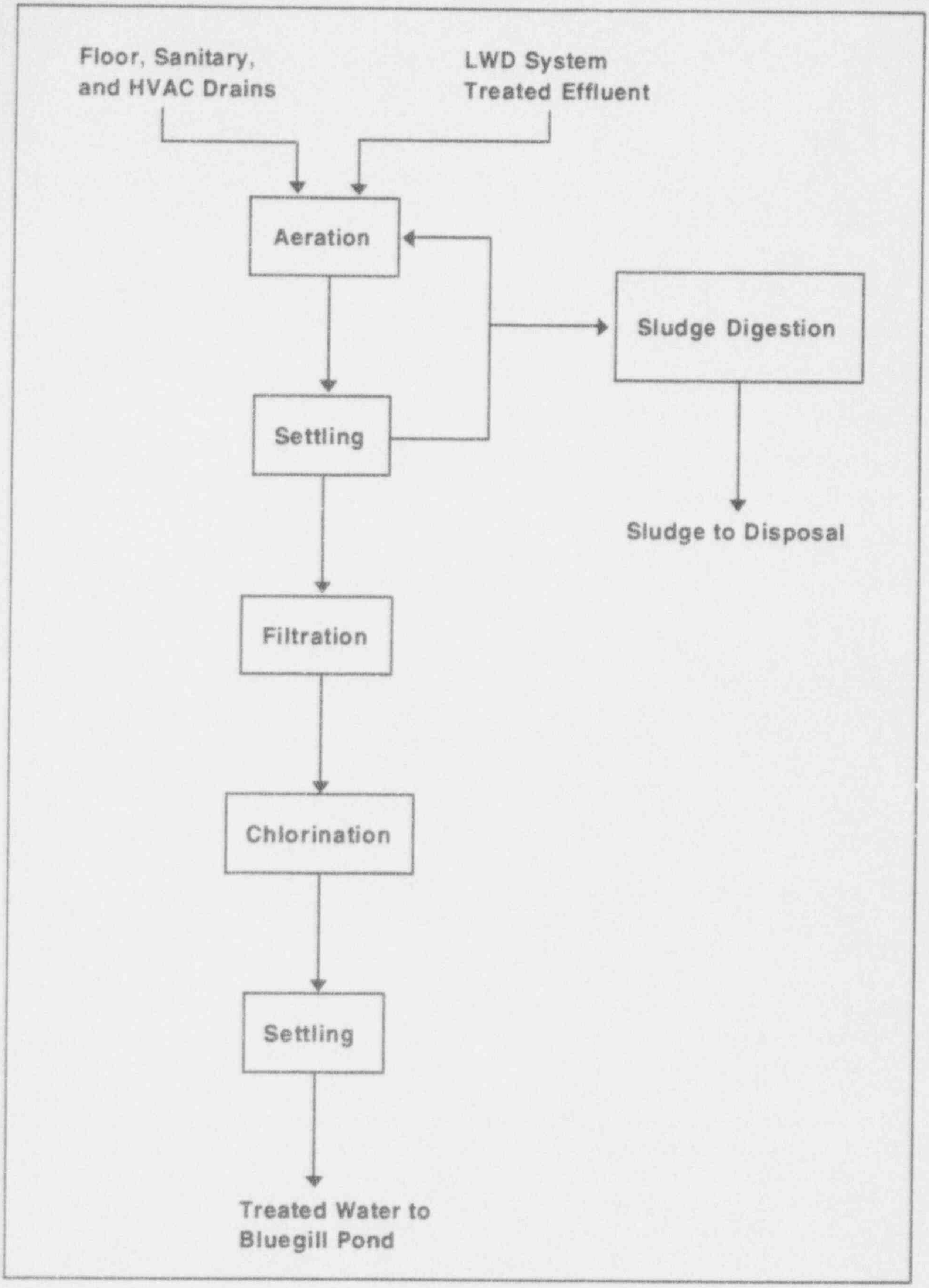

Figure 7.2 Schematic of the sewage treatment system

Table 7.2 Estimates of solid radioactive and mixed waste generated at CEC annually during operation (LES, 1993b)

Waste Type	Radiological Waste		Mixed Waste	
	Quantity (kg)	Uranium (kg)	Quantity (kg)	Uranium (kg)
Activated Carbon	680	55	45	0.5
Activated Alumina	160	1.8	-	-
Ventilation Filters	840	0.5	-	-
Demineralizers Resin	136	0.01	-	-
Waste Precipitate	200	36	-	-
Dryer Concentrate	1820	1.0	-	-
Solvent Recovery Sludge	-	-	115	5
Laboratory Wastes[a]	115	1.8	70	1.8
Trash	7270	10	230	0.5
Scrap Metal	130	trace	-	-
Fomblin Oil Recovery Sludge	25	0.5	-	-

[a] Dry wastes are in kg. Mixed waste includes water (60 percent), common non-hazardous laboratory chemicals, uranium, and small amounts of hazardous chemicals. These chemicals include isopropylether (60 kg), carbon tetrachloride (9 kg), carbon disulfide (2 kg), chromium compounds (0.5 kg), acetone (0.5 kg), and traces of n-hexane and 1,1,2 trifluoro-1,2,2 trichloroethane.

Miscellaneous waste includes paper, packing, and cleaning materials, and is segregated at the point of generation into radioactive, mixed, hazardous, and industrial waste containers. Plastic bags of each type of trash are transferred to the radioactive waste storage area for inspection for radioactive contamination. Radiologically contaminated, mixed, and hazardous waste are stored before being transported offsite to an authorized facility. Adsorbents and filters are produced in the treatment of gaseous effluents of autoclaves and desublimers, and in the operation of the general ventilation system. These materials are classified as radioactive, mixed, hazardous, or industrial waste on the basis of process knowledge and are handled in a manner like that used for miscellaneous solids. Sludges generated in the treatment of aqueous and nonaqueous liquid effluent streams are classified on the basis of process knowledge, and are collected, labeled, stored, and disposed of in a manner consistent with the hazard classification. Onsite treatment of precipitates and sludges is limited to dewatering by drainage or centrifugation.

The final category of CEC solid effluent is UF_6 tails from the separation cascades. Annual production of this material is 304 cylinders, each containing 12,700 kg (14 tons). The cylinders are stored onsite for a maximum of 15 years. Detailed discussion of the disposition of UF_6 tails is presented in SER Section 15.2.1.2. For the purposes of SER evaluations, it is assumed that the UF_6 tails will be converted to triuranium octoxide U_3O_8 and disposed in an appropriate facility, for example, a deep mine cavity. LES has committed in its license conditions to periodic inspections of stored cylinders so that any deterioration will be identified and remedied.

7.3.2 NRC Staff Evaluation of Solid Waste Management Systems

On the basis of this analysis, the NRC staff finds that the management of solid radiological and nonradiological wastes related to CEC operation is consistent with the ANPR and accepted health standards, and does not pose an undue risk to public health and safety.

7.4 Effluents

Each of the CEC waste management systems described in this section discharges material to the environment. This section summarizes the effluent characteristics and discharge rates used in analyzing the effects of normal operations.

Gaseous effluents containing radioactive material are released to the atmosphere at an elevation of 36.6 meters through three stacks adjacent to the Separations Building. Experience at the Urenco plants in Europe indicates that the release of uranium isotopes will be less than 4.4×10^6 Bq/yr (120 μCi/yr), and process-analysis-based estimates for the same releases are less than 1.1×10^6 Bq/yr (30 μCi/yr). A concentration-based release level has been proposed for the CEC at one-twentieth of the 10 CFR Part 20 limit for unrestricted areas. This release level is applied at the release point to ensure compliance with the 10 CFR Part 20 limit in the unrestricted area. This concentration-based guideline is equivalent to a release of 4.4×10^6 Bq/yr (120 μCi/yr). The NRC staff finds that 4.4×10^6 Bq/yr (120 μCi/yr) is a reasonable estimate of the upper bound of the annual release rate for CEC operation.

Liquid effluents containing uranium are released to Bluegill Pond from the CEC liquid waste disposal and sewage treatment systems. Experience at the Urenco plants in Europe indicates that this release will be in the range of 7.4×10^5 to 1.1×10^6 Bq/yr (20 to 30 μCi/yr). The NRC staff's analysis of the CEC treatment system feed streams and process steps including precipitation and evaporation with optional ion exchange indicates that releases within the proposed range will be likely. In addition, the applicant proposes that release to Bluegill Pond will be limited to less than 0.5 percent of the 10 CFR Part 20 limit for release to unrestricted areas. This commitment is equivalent to a release of 0.5×10^6 Bq/yr (14 μCi/yr) in the liquid effluent. For the purpose of impact analysis, the NRC staff adopted 1.0×10^6 Bq/yr (28 μCi/yr) as a conservative estimate of the upper limit for the annual release rate of uranium isotopes to Bluegill Pond. This release rate is conservative because it is 2 times the CEC action level for uranium in liquid effluent.

7.5 Effluent Monitoring and Sampling

This section explains the types of proposed effluent monitoring to indicate leaks in the process system by sampling and monitoring gaseous and liquid effluents from the CEC and evaluates the proposed CEC monitoring programs to determine their adequacy. CEC SAR Sections 6.4 and 8.3, and Proposed License Conditions (LES, 1993e) detail the proposed effluent control and monitoring systems.

7.5.1 Monitoring Gaseous Releases

LES proposes that the CEC gaseous effluent releases be monitored by using HF and particulate samplers. UF_6 hydrolyzes in atmospheric water vapor to form several byproducts, one of which is HF. The GEVS and a portion of the TSA HVAC system handle potentially contaminated air. HF monitors will be located in the GEVS, upstream of air filters, where there is a potential for UF_6 release to the atmosphere. For the potentially contaminated portions of the TSA HVAC system, the HF monitors are located in ventilation lines downstream of the HEPA filters. The applicant estimates that these monitors are capable of detecting 0.5 parts of HF per million. Because air exhaust systems operate continuously during normal and abnormal operations, the GEVS monitor will continuously monitor any gaseous HF releases (accompanied by uranium as UO_2F_2) which might enter the ventilation exhaust ducts.

Each of the three Separations Building stacks has passive, continuous samplers (isokinetically sampled from the air streams with no alarm capability) for collecting radioactive particles of uranium carried in the exhaust effluent air. The filters from the samplers provide air samples to be collected and counted (gross alpha) at least weekly; the counting instrument is capable of detecting a small fraction of the 10 CFR Part 20 limits for gaseous effluents.

Activated carbon or alumina filters and high efficiency particulate air (HEPA) filters in the GEVS and TSA ventilation systems remove reactive chemicals and radioactive particulates. The carbon filters are rated at 99-percent efficiency for HF removal, and the HEPAs are rated at 99.97-percent efficiency or better for removing 0.3 micrometer particulates. Air samplers placed after such filters serve as a check of filter operability and provide a record of effluent releases to the atmosphere during normal or abnormal operations. LES also commits to collecting and analyzing the isokinetic filters immediately if a GEVS HF monitor alarms, and perform daily analyses until the alpha levels in effluent return to normal (LES, 1993e). LES will also perform a running quarterly average for the weekly filters to be trended for comparison to action levels (proposed to be 4.0, 0.5, and 0.5 percent of the 10 CFR Part 20 limits for Stacks 1, 2, and 3, respectively). Should these limits be exceeded, LES is committed to perform isotopic analyses of the releases (LES, 1993e). If the action levels are exceeded for the weekly filter analysis, an investigation will be performed to determine the source of the release.

The proposed lower limit of detection (LLD) for gross alpha ($3.7x10^{-11}$ Bq/ml or $1x10^{-15}$ µCi/ml) will be 0.03 percent of the 10 CFR Part 20 limits for uranium, while the LLD for alpha isotopic analysis ($3.7x10^{-12}$ Bq/ml or $1x10^{-16}$ µCi/ml) will be 0.003 percent of the 10 CFR Part 20 limits for Uranium-234, -235, and -238 (Class D). The proposed action level for CEC Unit 1 would be $4.4x10^{-9}$ Bq/ml ($1.2x10^{-13}$ µCi/ml), and the action levels for Units 2 and 3 would be $5.6x10^{-10}$ Bq/ml ($1.5x10^{-14}$ µCi/ml). Unit 1 requires a higher action level because effluents from the GEVS and TSA areas will exhaust only through that stack. The action levels would be 4 percent of the 10 CFR Part 20 limits for Unit 1, and 0.5 percent of the limits for Units 2 and 3. Effluent releases are greatly diluted before reaching the site boundary where the diluted plume is also sampled as part of the environmental monitoring program. The NRC staff believes that this effluent monitoring information will be useful to CEC operators in detecting trends that could provide an early warning of potential failures in process confinement and treatment systems. Weekly filter samples be collected and composited quarterly. The applicant's PLC (LES, 1993e) specify that the LLD for the semi-annual composite analysis will not exceed $3.7x10^{-13}$ Bq/ml ($1x10^{-17}$ µCi/ml; that is, near normal background). The NRC staff finds the proposed uranium LLDs, action and administrative levels acceptable for ensuring protection of the public and the environment.

As proposed, alpha-in-air monitors in the UF_6 Handling and Blending Areas and HF air monitors in the ventilation systems of potentially contaminated areas of the TSA have alarm capability. If a predetermined setpoint is reached, the monitor activates an alarm to alert operators to the presence of uranium or HF (accompanied by radioactive particles) in building effluent air. This feature is very important for the Separations Building, the primary source of effluents, because 90 percent of the air collected in the HVAC exhaust ducts is normally recycled to the building supply air, and only 10 percent of the exhaust is released to the environment. If this recycled air were contaminated, it would present a risk to workers.

For the serious accidents considered in SAR Section 4.2 or SER Section 11, the CEC intends to isolate the room in the Separations Building in which an accident occurs by shutting down its ventilation system. However, even when the ventilation flow in this room stops, some leakage into the ventilation system is possible because of the negative pressure created by the continued operation of the ventilation systems in other rooms. That is, because each room has its own fan, air flow from the rest of the building up the stack continues and thereby creates a small negative pressure in the ventilation system of the shutdown room. This slight negative pressure in the accident area is desirable because it impedes leakage of radioactivity from the contaminated area into uncontaminated areas of the facility. The small difference in atmospheric pressure creates a small flowpath from the room up the stack unless a leak-tight damper is closed to totally isolate the room.

Effluent monitors in the shutdown section of the ventilation system are unable to give a quantitative estimate of the release rate because the flowrate would be unknown. However, the slightly negative pressure in the shutdown ventline means that airflow would be into the room containing the accident; through the ventilation system, where it would be filtered; and up the stack, where it could be precisely measured.

7.5.2 Monitoring and Sampling Liquid Effluent Releases

Details of the proposed treatment of liquid wastes generated by normal operations are discussed in the LES SAR, ER, and the National Pollutant Discharge Elimination System (NPDES) Application to Environmental Protection Agency (EPA) and Louisiana (Attachment 2; LES, 1992d). A summary of the proposed radioactive liquid effluent treatment system is shown in Figure 7.1. The principal method of treatment is evaporating water from effluents to yield a dry powder of dissolved and suspended solids. The solids are disposed of offsite as low-level waste. The evaporated water is recondensed, collected, and sampled. If the concentrations are below the administrative limit (5 percent of the concentration limit in 10 CFR Part 20, Appendix B), the liquids are transferred to the Sewage Treatment System. Liquid effluent of the Sewage Treatment System is sampled and released to Bluegill Pond if the water quality meets EPA Water Quality Standards and Criteria, and the requirements of the Louisiana Administrative Code and 0.5 percent of the limits in 10 CFR Part 20, Appendix B.

All potentially contaminated liquid effluents generated onsite, primarily in the Separations Building, are collected in one of several waste tanks before treatment and release to the Sewage Treatment System. The discussion here is general; SAR Sections 6.4.14 and 8.3.3 provide additional details. A summary of the liquid effluent system and flow paths is shown in Figure 7.1.

Liquids from floor, laboratory, and sink drains are collected in their own waste tanks, as are liquids from the laundry and decontamination facilities. When a tank is full, the contents are sampled and analyzed for radioactivity. Liquids are transferred to the LWDS, where they are treated to remove uranium and chemical contaminants. This monitoring provides assurance that the concentrations of uranium at the point of release to the sewage treatment flow are well below the concentration limits of 10 CFR Part 20 Appendix B, Table 2, and that the water quality meets the requirements of the EPA and the State.

Liquid effluents from building roof and yard drains, and stormwater runoff from roadways and parking lots flow to the Hold-Up Basin (prior to plant operation), where liquids are to be periodically sampled and analyzed to ensure that low-level contaminants are well below any regulatory limits. After start-up, the Hold-Up Basin does not remain functional, but liquids continue to pass through it and discharge to Bluegill Pond. The Proposed License Conditions (PLC) state that the CEC will sample stormwater runoff consistent with requirements of the State for gross alpha analysis, and if such runoff exceeds 0.74 Bq/liter (20 pCi/liter) above background, the source will be investigated and corrective action will be taken.

7.5.2.1 Monitoring and Sampling Radioactive Liquid Effluents

All routine releases of LWDS liquid effluents are batch releases which are sampled before release, and all liquids leaving the site are added to the normal continuous flow of sewage treatment water. This flow is continuously sampled downstream in order to provide a composite sample for quarterly analysis.

The applicant proposes to set administrative limits at 0.5 percent of the new limits of 10 CFR 20.1302(2) for uranium in routine liquid effluents from the sewage treatment system before release to Bluegill Pond. LES commits to providing sufficient sensitivity and reliability to promptly detect uranium releases (LLD of 5.5×10^{-6} Bq/ml or 1.5×10^{-10} μCi/ml). In addition, LES will composite samples semi-annually and subject them to state-of-the-art laboratory measurements capable of detecting uranium concentrations on the order of background. If the 0.5 percent Action Level is reached, the CEC Manager and Compliance Superintendent are notified and the cause investigated, corrected and documented. The NRC staff finds that CEC's proposed LLD for routine gross alpha analyses (ten times background) to be acceptable, because water is also sampled after dilution of liquid releases in Bluegill Pond as part of the environmental monitoring program capable of measuring background levels of uranium.

An accidental release from the failure of a single liquid waste line or tank is unlikely to reach the sewage effluents because of the series of holding tanks in the liquid effluent treatment system. For example, a release from a broken line or an overflowing tank would most likely flow to a floor drain which would flow to another holding tank.

7.5.2.2 Monitoring and Sampling Chemical Liquid Effluents

The proposed monitoring and sampling program for chemicals in liquid effluents is described in the Louisiana Water Discharge System Permit (LWDSP) Application to the State of Louisiana (LES, 1992d). The NRC's position is that nonradiological liquid effluents, though not within its regulatory purview, must meet appropriate federal and state standards. Proposed regulatory and administrative limits for chemicals in sewage treatment discharge water are presented in the LWDSP Permit Application, Attachment 8. To demonstrate compliance with these standards, LES can be required to report analytical results of tests of representative samples collected periodically. LES commits to meeting these standards and will be required to do so in order to receive the necessary approvals for nonradiological discharges. When these standards are met, no significant impacts are likely to occur. To ensure that the receiving onsite and offsite waters are not affected by the liquid effluents, CEC also conducts an operational environmental monitoring program at the site and nearby locations (see DEIS, Sections 7.1.3 and 7.2.4) (NRC, 1993a).

Most liquid wastes which could contain measurable concentrations of chemicals are treated by evaporation and other means to remove chemicals and uranium, and to reduce the remaining liquid wastes to solids for traditional solid waste disposal. Those liquids which can be recovered and recycled (e.g., Fomblin oil) are reused to reduce waste production and costs. Nonaqueous liquid wastes (e.g., lubrication oils, solvents) are disposed off in accordance with appropriate federal and state requirements.

7.5.3 Regulatory Requirements for Effluent Monitoring and Sampling

The ANPR requires the capability to measure liquid and gaseous effluent releases during normal operations and under accident conditions. Because such measurements can be made by several independent systems, the requirements are discussed further below and used to evaluate CEC's proposed effluent monitoring and sampling programs.

7.5.3.1 Radiological Gaseous Effluent Monitoring Program

The ANPR states the NRC requirements for monitoring radioactive effluents from the proposed CEC facility. For gaseous discharges, the NRC requires that an appropriate means be used to measure the amount of radioactivity in gaseous effluents during normal operations and under accident conditions, and the flow rate of the diluting medium (that is, air). The capability of the CEC effluent monitoring system to measure the entire range of releases for normal operations and severe accidents is specified as a license condition (LES, 1993e). In addition, the proposed system measures the flowrate in the exhaust stack and the concentrations of uranium in the effluents.

CEC has committed to providing monitoring capabilities consistent with the NRC requirements (LES, 1993e). Applicant's PLC specify that the semi-annual composite measurements provide statistically meaningful measurement of uranium down to near background levels, which are on the order of 3.7×10^{-13} Bq/ml (1.0×10^{-17} µCi/ml), as discussed in NCRP Report 94.

For gaseous effluents from normal operations or abnormal conditions, CEC uses alpha continuous air monitors (CAM) and HF monitors to give an alarm for radioactivity. This system measures routine and abnormal releases. By promptly alerting operators of potentially serious radioactivity releases, the system initiates steps to mitigate abnormal releases. The NRC staff concludes that the system satisfies the requirements of the ANPR for routine and abnormal releases from the proposed facility.

Accident release estimates are based on collected passive, particulate, effluent stack samples, with alarm capability of gaseous uranium entering ventilation ducts provided by HF monitors or alpha-in-air monitors in areas where accidental releases are possible. The effluent monitoring systems of the CEC facility are not classified in Section 4.2 as structures, systems, or components important to safety because separate, redundant Class I instrumentation and control systems terminate accidental releases (automatic or operator-initiated). In addition, multiple effluent monitors (uranium and HF) in work areas and ventilation systems give adequate assurance that any accidental releases are detected. Although the operation of an effluent monitoring system during a serious accident is not important to safety, the ANPR requires that an applicant measure effluent releases during severe accidents. Thus, in-plant monitoring systems must be operable to give adequate assurance that there is no undue risk to public health and safety. If the proposed system cannot provide such assurance, an alternative measurement system must be available to do so. Assurance that releases during serious accidents do not compromise safety could also come from other means such as air sampling in the vicinity of the stack and at the site boundary during an accidental release from the facility.

Although ventilation monitors provide an indication of a radioactive release from an accident area, the quantity of the release cannot be estimated because the leakrate cannot be known. However, any leakage would be collected by the balance of the operating system and measured in stack effluents, as required by the ANPR. On the basis of LES commitments and the CEC capabilities presented, the NRC staff concludes that the CEC facility can comply with the ANPR requirements to monitor quantitatively any accidental releases by one or more of the proposed monitoring systems.

7.5.3.2 Radiological Liquid Effluent Monitoring

The ANPR requirements for sampling and measuring radioactivity also apply to monitoring liquid effluents from the CEC facility. Thus, monitoring CEC liquid effluents must determine their concentrations of radioactivity and volumes.

Because, under normal operating conditions, the concentrations of radioactivity in such effluents are very low, LES must ensure that the sensitivity of CEC monitoring of operational effluent is comparable to the high sensitivity of its environmental monitoring program. High sensitivity detects small changes in normal, above-background concentrations which can give early warning of impending system problems before they become serious enough to cause costly plant shutdowns. Thus, high-sensitivity monitoring of liquid effluent can be cost-effective over the long term.

LES has proposed that the administrative limit for routine concentrations in liquid effluents be 0.5 percent of the 10 CFR Part 20 limit at the point of release (i.e., before dilution in the receiving water). Measuring small concentrations is done daily by taking a composite sample and making a single gross alpha/beta count on a proportional counter. These daily measurements can detect a small fraction of the concentration limits in 10 CFR Part 20, Appendix B, Table 2, (restricts ingestion CEDEs to 0.5 millisieverts or 50 mrem/yr). However, in order to demonstrate that the EPA Clean Water Act limits (CEDE of 4 mrem, or 0.04 mSv/yr) are not exceeded, it will also be necessary to composite representative effluent samples for each quarter. The composite sample must be subjected to a more sensitive, state-of-the-art laboratory analysis capable of detecting background levels of uranium. The average background level of uranium in U.S. surface waters is on the order of 3.7×10^{-6} Bq/ml (1E-10 μCi/ml) (NCRP Report 94, 1987). LES commits (LES, 1993e) to a program which satisfies these requirements.

If an accident resulting in a serious liquid effluent release (e.g., during a design basis earthquake) were to occur, more frequent sampling (e.g., daily or hourly) and simple gross alpha and beta counting to detect wide-ranging concentrations (from those below NRC release limits to those orders of magnitude above them) are done easily, with no requirement for high sensitivity. Although such requirements are specified in emergency procedures, the technology is simple and presents no problem in implementation. Thus, the NRC staff concludes that the CEC liquid effluent monitoring programs satisfy ANPR radiological liquid effluent monitoring requirements for both routine and accidental releases.

7.5.3.3 Chemical Effluent Monitoring Program

The gaseous chemical effluent of major concern (Section 5.5.1) is HF produced by the hydrolysis of UF_6. The uranium monitoring systems discussed in Section 7.5.3.1 indicate the potential presence of HF in the ventilation system. In addition, the HF detectors in the GEVS and potentially contaminated portion of the TSA HVAC system would signal release of HF to the atmosphere from these areas. If these monitors survive the effects of a serious (for example, a DBE) accident, they give the alarm and monitor chemical releases during post-accident recovery operations.

The simultaneous loss of the total inventory of all other onsite chemicals has no significant offsite consequences (SER Chapter 11). Thus, the NRC staff will not require that LES monitor these other chemicals in gaseous effluents from routine operations. However, EPA or the State of Louisiana may impose such requirements.

Because HF is the major gaseous chemical effluent of major concern, the NRC staff concludes that the proposed monitoring system adequately measures the chemical effluents expected from operation of the CEC facility.

7.6 Ensuring that Radioactivity Releases are ALARA

The applicant must ensure that normal operational releases are ALARA. It is expected that the combination of siting, design, and control features built into the plant as well as any administrative controls incorporated into its operating procedures result in public radiological effects which are only a small fraction of the 1.0 mSv (100 mrem) annual Total Effective Dose Equivalent (TEDE) limit specified in 10 CFR 20.1301. Other ALARA guidance is found in Regulatory Guides 4.14, 4.15, and 4.16; ANSI N13.1-1969; and ANSI N42.18.

Monitoring operational releases in facility effluents and measuring concentrations of the released radionuclides in the offsite environment can confirm that the ALARA goals have been met (10 CFR 20.1302). The CEC Draft Environmental Impact Statement (DEIS) (NRC, 1993a) addresses the operational environmental monitoring program. LES must provide assurances that potential accidents do not result in undue risk to the health and safety of the public. This section briefly addresses each of these issues.

7.6.1 Siting, Design, and Control Features

The ANPR requires that enrichment facilities be designed so that normal releases of radioactive materials are ALARA. ALARA guidance is discussed in a recent draft Regulatory

Guide ("'ALARA' Levels for Effluents from Materials Facilities", DG-8013, October 1992). The draft guide incorporates the elements of a Memorandum of Understanding (MOU) between the NRC and the EPA regarding nonreactor facility effluent controls that would restrict offsite dose to an ALARA goal of 0.1 mSv (10 mrem) (TEDE) per year of operation (57 FR 60778, December 22, 1992). To meet this goal, the design of structures, systems, and components to prevent releases and mitigate the consequences of accidents must be carefully evaluated to ensure that the proposed facility is capable of keeping offsite doses to the public ALARA. The details of the facility design are addressed in Section 4.0.

CEC SAR (LES, 1993a) Section 4.4 details the CEC design and control features which would protect the public from radiation during routine operations (pp. 4.4-1 to 4.4-14). These features are also discussed in SER Chapters 4 and 5. The more general and less detailed discussion here avoids unnecessary redundancy. In summary, the design and control features proposed for the CEC facility reflect years of successful operation for similar types of European facilities and, as a result, represent proven systems which have operated with acceptably low releases to the environment for several decades.

The principal design features which ensure that releases are ALARA are the primary confinement system (internationally approved cylinders and the process equipment) in which, with the exception of the autoclaving operation, all processes are conducted under negative pressure. This mode of operation ensures that leakage which might occur is to the process system rather than to the plant. The autoclave, in which the pressure is greater than ambient in order to force UF_6 gas from the storage cylinders into the process stream, itself provides secondary confinement. In the event of a pressure or temperature upset in the enrichment process, the UF_6 can also be diverted to the uranium tails and product take-off systems or to a contingency dump system in order to avoid potential damage to the centrifuges and releases to the environment.

When maintenance is required, piping and equipment are isolated, evacuated, and purged by portable vacuum pump sets which contain a layered, activated carbon, and aluminum oxide trap; an aluminum oxide trap; and an activated charcoal trap in order to remove any hazardous or radioactive materials. Any material not removed by the portable vacuum pump set is released to the GEVS for further treatment.

The TSA HVAC system for normal effluent control from work areas uses mobile enclosures around parts of the process system and equipment when maintenance is conducted. For those releases of uranium which might occur in spite of design features, the CEC facility intends to remove most of the HF and uranium before they can be released out a stack to the environment. Section 7.5.1 discusses the effectiveness of these systems.

UF_6 vented from cylinders is collected and cooled by a refrigerant in a desublimer subsystem. HF and uranium are removed by passing any remaining gases through a series of activated carbon and aluminum oxide (Al_2O_3) traps. Any traces of HF and uranium remaining are released to the GEVS for further treatment before release to the environment.

The major siting feature to help maintain offsite radiological effects to ALARA levels is the large size of the proposed site. Site size contributes to dilution of any gaseous or liquid releases before they reach the site boundary.

The first systems confirming that the facility is operating as designed are the HF and uranium monitors discussed in Section 7.5. Although not classified as important to safety, these systems can reliably monitor any releases during normal and abnormal conditions. Such measurements also detect unfavorable trends in releases and give timely notice to permit the causes to be corrected before the release rates become unacceptable. In addition, the HF alarm features of some of these systems can give operators prompt warning of UF_6 releases that might exceed the administrative limits so that immediate action can be taken to mitigate the releases.

Thus, the NRC staff finds that the combination of siting, design, and control features is adequate to meet all relevant ALARA requirements of the ANPR.

7.6.2 Administrative Controls

Administrative controls include requirements for construction material certification, fabrication procedures, structural analysis, stringent technical qualifications and training of operators and technical personnel, quality assurance, and documentation as well as administrative limits and implementing procedures to ensure that concentrations of HF and uranium in gaseous effluents at the point of release from the facility are small fractions of the release concentration limits in 10 CFR Part 20, Appendix B, Table 2. Meeting such limits gives reasonable assurance that mitigating actions can be taken quickly to prevent continuous gaseous releases from reaching concentrations in excess of 10 CFR Part 20 limits, even for short periods. Administrative limits for liquid effluents serve a similar function for batch releases of liquids to the environment. Procedures which require actions such as leak testing of process lines or evacuation and purging of process lines containing nitrogen before disconnecting them also help to minimize the potential for releases and achieve ALARA design objectives. The NRC staff finds that the proposed administrative controls for CEC operation give additional assurance that the facility will meet the ALARA requirements of the ANPR.

7.6.3 Effluent Monitoring Program

LES proposes gaseous and liquid effluent administrative limits set at 5.0 percent and 0.5 percent, respectively, of the NRC limits in 10 CFR Part 20, Appendix C, Table 2, at the point of release (Section 7.5.1). That is, the dose from inhaling air directly from the stack would be an annual CEDE of 25 microsieverts (2.5 mrem); the dose from ingesting water directly from liquid releases would be a CEDE of 2.5 microsieverts (0.25 mrem) per year. Because atmospheric releases would be diluted by factors on the order of a million before they reach the nearest resident, CEDEs to individuals in the offsite population would be only a small fraction of allowable doses from normal operational releases. Similarly, liquid releases would be diluted by orders of magnitude in surface waters or ground water and result in much lower offsite doses. For these reasons and the fact that the CEC facility reflects the latest design and technology for gaseous centrifuge enrichment, the NRC staff finds that the CEC facility can limit all effluent releases to ALARA.

8 RADIATION PROTECTION

The objective of the radiation protection program is to provide adequate protection of the Claiborne Enrichment Center (CEC) work force and the general public residing near the site under normal conditions of operation and following accidents. The following sections discuss the types of radioactivity and radiation that will be encountered at the site, the radiation protection programs for workers and the general public, and the NRC evaluation of these programs and potential impacts of CEC operation. A major source of information for this chapter is the CEC Safety Analysis Report (SAR), as revised (LES, 1993a).

8.1 Radiation and Radioactivity Sources

The predominant radioactive material to be utilized at the site will be natural, low-enriched, and depleted uranium primarily in the form of uranium hexafluoride (UF_6). However, other uranium compounds (for example, in wastes) will also be present as gases, liquids, and solids. Natural uranium is about 99.3 percent uranium-238 (U-238), and about 0.71 percent uranium-235 (U-235). The maximum enrichment proposed by the applicant, Louisiana Energy Services (LES), for CEC is 5.000 weight percent (wt %) U-235. In the depleted uranium tails, the U-235 content will be reduced to about 0.25 wt %. The numerous daughters, such as thorium and radium radionuclides, found accompanying natural uranium will be removed in the uranium milling process prior to being sent to the CEC. Incidental radioisotopes that accompany uranium are shorter-lived daughters (for example, thorium-234 [Th-234] and metastable protactinium-234 [Pa-234m]) which would "grow" into partial or complete radioactive equilibrium with long-lived parents. The decay chains for nuclides of interest at CEC are shown in Figure 8.1 and decay data are listed in Table 8.1. The proposed possession limits to be established by the license for special nuclear material and source material are approximately 1.5 million kilograms (3.3 million pounds) and 62.5 million kilograms (138 million pounds), respectively (LES, 1993e).

As can be seen from Table 8.1, most of the radiation emitted by the radioisotopes that would be encountered are alpha and beta particles which are non-penetrating forms of radiation that would be shielded from workers by UF_6 storage cylinders and primary containment systems (for example, process lines). Due to the high density of UF_6 when stored as a solid, the material would also provide considerable self-attenuation of x-rays and gamma rays from the uranium series nuclides present. A significant portion of the direct radiation encountered at the CEC will be in the form of bremsstrahlung radiation, which will be generated by the interaction of beta radiation with high atomic number atoms, such as uranium in UF_6 and, to a lesser extent, iron in UF_6 cylinders.

A primary concern for most LES operations would be incidental or accidental inhalation of uranium, which can cause non-stochastic chemical damage to the kidney (nephrotoxicity) if intakes exceed a threshold within a specified period of time. Significant releases of UF_6 to work areas are unlikely, since the entire centrifuge system, with the exception of the

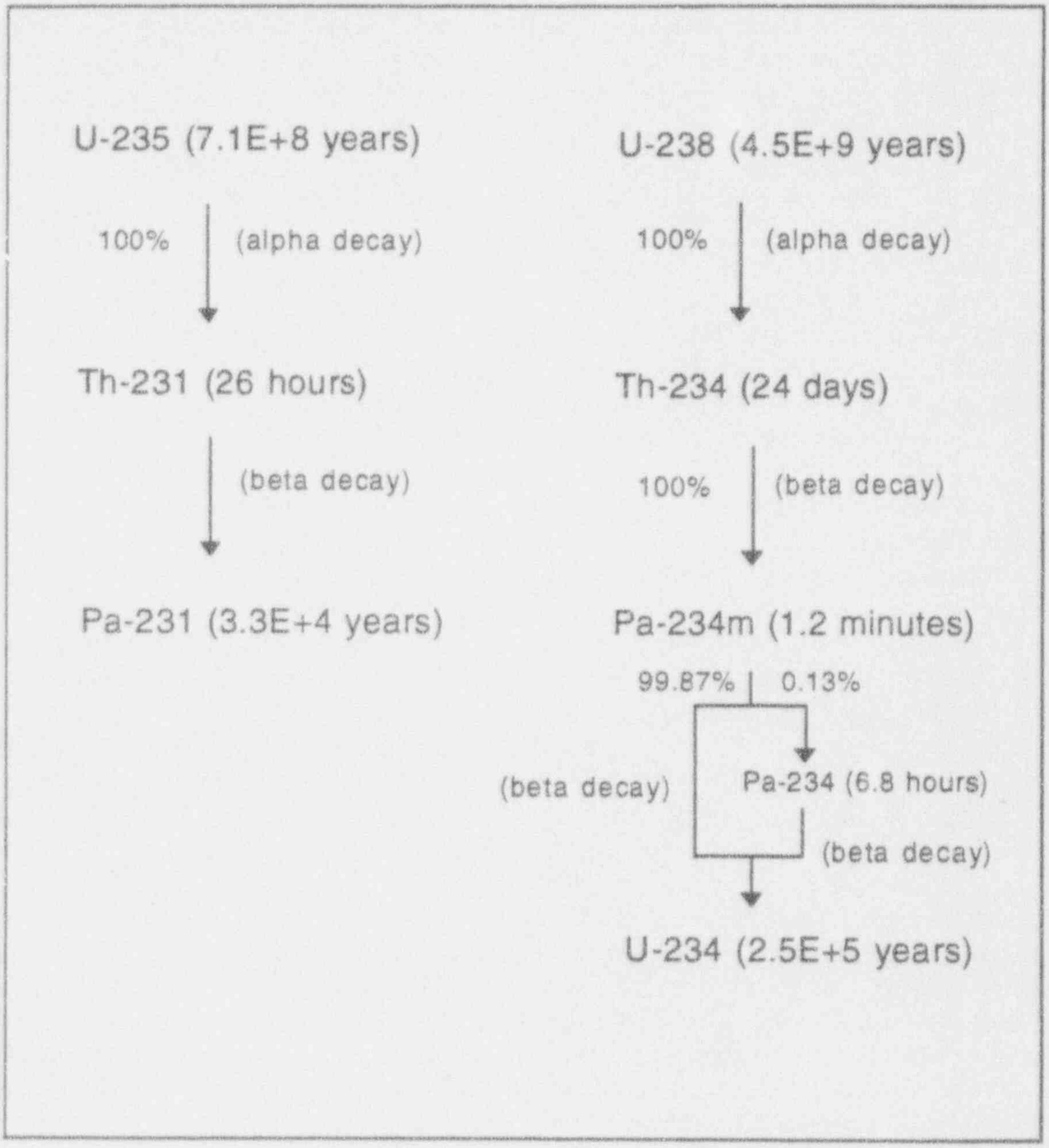

Figure 8.1 Uranium and decay products of interest at the CEC

Table 8.1 Radiological characteristics of natural UF_6 feed

Nuclide	Atom Ratio ppb in Natural Uranium	Half Life*	Maximum Radiation Energies (MeV) and Intensities		
			Alpha α	Beta β	Gamma γ
${}_{92}U^{238}$	9.927E8	4.51E9y	4.15 25% 4.20 75%	N/A	N/A
${}_{90}Th^{234}$	1.45E-2	2.41E1d	N/A	0.10 21% 0.19 79%	0.06 3.5% 0.09 4.0%
${}_{91}Pa^{234m}$	4.9E-7	1.17m	N/A	2.29 98%	0.77 0.3% 1.00 0.6%
${}_{92}U^{234}$	5.44E4	2.47E5y	4.72 28% 4.77 72%	N/A	0.05 0.2%
${}_{92}U^{235}$	7.205E6	7.1E8y	4.37 18% 4.40 57% 4.58 8%	N/A	0.143 11% 0.185 54% 0.204 5%

* y = years, m = minutes, d = days; adopted from Table 3.2-2 (LES, 1993b)

autoclaves, is to be operated in a partial vacuum so that leaks are into the system, not into the work areas. Most of the other sources of radioactivity utilized at the facility would be small calibration and radiochemistry (quality control) standards which pose little radiation exposure risk to workers and none to the public. The proposed byproduct possession limits and uses are summarized in Table 8.2 (LES, 1993e). Byproduct material may be in solid, liquid, or gaseous form, and is not necessarily restricted to sealed sources.

8.2 Radiation Protection Design Features

Section 20.1701 of Title 10 of the Code of Federal Regulations (CFR) (NRC, 1991c) requires licensees to "use, to the extent practicable, process or other engineering controls (for example, containment or ventilation) to control the concentrations of radioactive materials in air." To control worker exposure, the Advance Notice for Proposed Rulemaking (ANPR) (NRC, 1988a) requires that "radiation protection systems be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials." In addition, "structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure, must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel" (NRC, 1988a).

Table 8.2 Proposed possession limits for byproduct material (LES, 1993e)

RADIONUCLIDE	CURIE LIMIT	USE
H-3	1.0	Instrument calibration and/or quality control
C-14	0.5	Instrument calibration and/or quality control
Cr-51	0.1	Instrument calibration and/or quality control
Mn-54	0.1	Instrument calibration and/or quality control
Fe-55	0.1	Instrument calibration and/or quality control
Fe-59	0.1	Instrument calibration and/or quality control
Co-57	0.1	Instrument calibration and/or quality control
Co-58	0.1	Instrument calibration and/or quality control
Co-60 (sealed only)	0.1	Instrument calibration and/or quality control
Co-60 (any form)	0.02	Instrument calibration and/or quality control
Ni-63	0.25	Instrument calibration and/or quality control and/or internal instrument standard
Ni-65	0.25	Instrument calibration and/or quality control and/or internal instrument standard
Zn-65	0.1	Instrument calibration and/or quality control
Sr-89	0.1	Instrument calibration and/or quality control
Sr-90	0.1	Instrument calibration and/or quality control
Y-90	0.1	Instrument calibration and/or quality control
Tc-99*	1.1	(See note *)
Ag-110m	0.1	Instrument calibration and/or quality control
Cd-115m	0.1	Instrument calibration and/or quality control
I-131	0.1	Instrument calibration and/or quality control
Ba-133	0.25	Instrument calibration and/or quality control and/or internal instrument standard
Cs-134	0.3	Instrument calibration and/or quality control and/or internal instrument standard
Cs-137 (sealed only)	0.3	Instrument calibration and/or quality control and/or internal instrument standard

Table 8.2 Proposed possession limits for byproduct material (LES, 1993e) (continued)

RADIONUCLIDE	CURIE LIMIT	USE
Cs-137 (any form)	0.05	Instrument calibration and/or quality control and/or internal instrument standard
Eu-152	0.1	Instrument calibration and/or quality control
Ra-224	0.1	Instrument calibration and/or quality control
Ra-226	0.1	Instrument calibration and/or quality control
Ra-228	0.1	Instrument calibration and/or quality control
Ac-226	0.1	Instrument calibration and/or quality control
Ac-227	0.1	Instrument calibration and/or quality control
Ac-228	0.1	Instrument calibration and/or quality control
Th-228	0.1	Instrument calibration and/or quality control
Th-229	0.1	Instrument calibration and/or quality control
Th-230	0.1	Instrument calibration and/or quality control
Th-232	0.1	Instrument calibration and/or quality control
Th-234	0.1	Instrument calibration and/or quality control
U-233	0.1	Instrument calibration and/or quality control
U-234	0.1	Instrument calibration and/or quality control
U-235	0.1	Instrument calibration and/or quality control
U-236	0.1	Instrument calibration and/or quality control
U-238	0.1	Instrument calibration and/or quality control
Pa-231	0.1	Instrument calibration and/or quality control
Pa-232	0.1	Instrument calibration and/or quality control
Np-234	0.1	Instrument calibration and/or quality control
Np-235	0.1	Instrument calibration and/or quality control
Am-241	0.1	Instrument calibration and/or quality control

*Tc-99 shall not exceed 0.001 micrograms per gram of total uranium in accordance with ASTM Standard Specification for Uranium Hexafluoride for Enrichment (C 787-90). (This is included only to address trace impurities in UF_6 containers.)

The CEC SAR identifies considerations related to exposure control provided for in the CEC design which include:

- confinement of radioactive materials in cylinders, lines, tanks and process systems

- HVAC and gaseous effluent ventilation system designs that remove particulate radioactivity from work areas or remove potential sources of radioactivity, treating them and releasing the cleaned effluent outside the facility

- use of "elephant trunks" for removal of particulate radioactivity from worker breathing zones during maintenance, connection and disconnection of UF_6 cylinders, etc.

LES has committed to designing the CEC facility ventilation equipment such that normal air flow will generally be from areas of lesser potential contamination to areas of higher potential contamination (LES, 1993e).

In addition to exposure control considerations, the ANPR requires that radiological alarm systems be designed with provisions for calibration and testing for operability. Effluent monitoring systems must be provided, and their designs must provide for measurement of releases during normal and accident conditions. Thus, the systems must measure both concentrations of radioactivity in effluent and the flow rate of the effluent streams. Direct radiation monitoring systems must be provided in any direct radiation areas. Finally, the design of enrichment facilities must facilitate decommissioning and removal of equipment and radioactive wastes at the end of plant life, which would reduce occupational doses as well.

The NRC staff has reviewed the available information on these design related aspects of the CEC radiation protection systems, and concludes that they are consistent with the general requirements of 10 CFR Part 20 and the ANPR guidance.

8.3 Radiation Protection Program

The CEC radiation protection program involves the entire range of facility operations which could affect worker or public safety in normal operations or during accident conditions (See also Sections 10.8, "Emergency Planning" and 11.3, "Accident Analysis").

The applicant states in Section 8 of the SAR that the CEC Manager will be responsible for the protection of all persons against radiation exposure resulting from facility operations and materials, and for compliance with applicable NRC regulations. In addition, the Operations Superintendent will be responsible for operating the facility safely and in accordance with procedures so that all effluents released to the environment and all exposures to the public and CEC personnel meet limits specified in applicable regulations, procedures and guidance (LES, 1993e). The Health Physics Manager and Health Physics Staff will be responsible for:

- establishing and carrying out the radiation protection program
- establishing and maintaining an ALARA program and ensuring that ALARA is practiced by all personnel
- adequately staffing the radiation protection program with qualified personnel
- preparing and maintaining procedures associated with the program
- maintaining and calibrating radiation measurement systems and instruments, including verification of required lower limits of detection (LLD) or alarm levels
- modifying the program based upon experience and facility history
- establishing and maintaining the radiological environmental monitoring program
- reviewing and auditing the efficacy of the program in complying with NRC and other governmental regulations and applicable Regulatory Guides
- establishing and maintaining a respirator usage program
- monitoring worker doses, both internal and external
- complying with the radioactive materials license for the facility
- handling of radioactive wastes when disposal is needed
- performing audits of the radiation program on an annual basis
- posting Radiation Control Areas (RCA) and Radiation Control Zones (RCZ), as appropriate
- developing occupancy guidelines, if needed
- establishing and maintaining a radiation safety training program for personnel working in radiation control areas or zones
- preparing an ALARA report annually (LES, 1993a).

The Health Physics Manager or designee will also be responsible for determining the need for, issuing, and closing out of Radiation Work Permits (RWPs). The Health Physics Manager or designee shall review planned activities, or changes in activities inside radiation control areas, or activities involving licensed material, for potential for causing radiation exposures to exceed action levels and radioactive contamination (LES, 1993e).

The Health Physics Manager will have direct access to the CEC Manager to resolve radiation safety problems (LES, 1993e).

During emergency conditions the Health Physics Manager's duties shall also include:

- providing Emergency Operations Center personnel information and recommendations concerning chemical and radiation levels at the facility

- gathering and compiling onsite and offsite radiological and chemical monitoring data

- recommending actions deemed necessary for limiting exposures to facility personnel and members of the general public

- performing decontamination activities (LES, 1993e).

CEC Quality Assurance Group personnel and other individuals technically qualified to perform audits and inspections, shall be responsible for inspecting (routinely) and auditing (at least annually) the efficacy of the program in complying with written procedures, license conditions, and NRC and other governmental regulations (LES, 1993e).

The applicant states in Section 8 of the SAR that personnel whose duties require (1) working with radioactive material, (2) entering radiation controlled areas, (3) controlling facility operations that could affect effluent releases, or (4) directing the activities of others, will be trained such that they understand their responsibilities (LES, 1993a).

The CEC radiation protection program is committed to the philosophy of "As Low As is Reasonably Achievable" (ALARA) for all operations involving source, byproduct, and special nuclear material. LES has committed to training all personnel in ALARA concepts, and is committed to ALARA principles in the establishment of the radiation protection program, including establishment of administrative limits (that is, licensee established limits that are below the NRC regulatory limits and whose purpose is to achieve an effective ALARA program) for minimizing occupational exposures, operational procedures, work plans, dosimetry, survey and monitoring programs, and equipment (LES, 1993a). LES has committed to preparing an annual ALARA report which will cover radiological exposure and effluent release data for trends, audits and inspections, and the use, maintenance and surveillance of equipment for exposure and effluent control. The HP Manager will be responsible for preparing the report, and submitting it to the CEC Manager and the Facility Safety Review Committee (LES, 1993e). The NRC staff concludes that the responsibilities assigned to CEC personnel and the extent of incorporation of the ALARA principle in CEC's radiation protection program are consistent with the requirements of 10 CFR Part 20 and the ANPR guidance.

8.4 Occupational Radiation Protection

LES proposes to define radiation areas (RCA/RCZ) for protection of workers from radiation and from the chemical toxicity of uranium (see proposed CEC radiological access zones shown in Figure 8.2) as follows (LES, 1993e):

An RCA is:

1) An area where airborne concentrations or radionuclides (corrected for background) are sufficient to have the area designated as an "Airborne Radioactivity Area" as defined in 10 CFR 20.1003, or

(2) An area where the radiation levels (corrected for background) are sufficient to have the area designated as a "Radiation Area" as defined in 10 CFR 20.1003, or

(3) An area where the contamination levels (sum of fixed and removable, corrected for background) exceed 150 dpm/100 cm^2 alpha or beta/gamma, or

(4) An area where the intake of soluble uranium following a 40-hour exposure in one week is likely to reach 1 milligram.

An RCZ is:

(1) An area where airborne concentrations of radionuclides (corrected for background) are sufficient to have the area designated as an "Airborne Radioactivity Area" as defined in 10 CFR 20.1003 and will result in a Committed Effective Dose Equivalent (CEDE) that is greater than 25 percent of the annual organ or total body 10 CFR Part 20 limit if respiratory protection is not utilized, or

(2) An area where the radiation levels (corrected for background) are sufficient to have the area designated as a "High Radiation Area" as defined in 10 CFR 20.1003. Small areas within an RCA that meet the definition of an RCZ may be posted without having the entire area designated as an RCZ, or

(3) An area where the removable contamination levels (corrected for background) exceed 1,000 dpm/100 cm^2 alpha or beta/gamma. This would apply only to areas that are accessible to workers when no work intrusive to facility components is being performed. Small areas not accessed by workers and areas not accessible to workers may be posted without having the entire area designated as an RCZ, or

(4) An area where the intake of soluble uranium following a 40-hour exposure in one week is likely to exceed 1 milligram without the use of respirators.

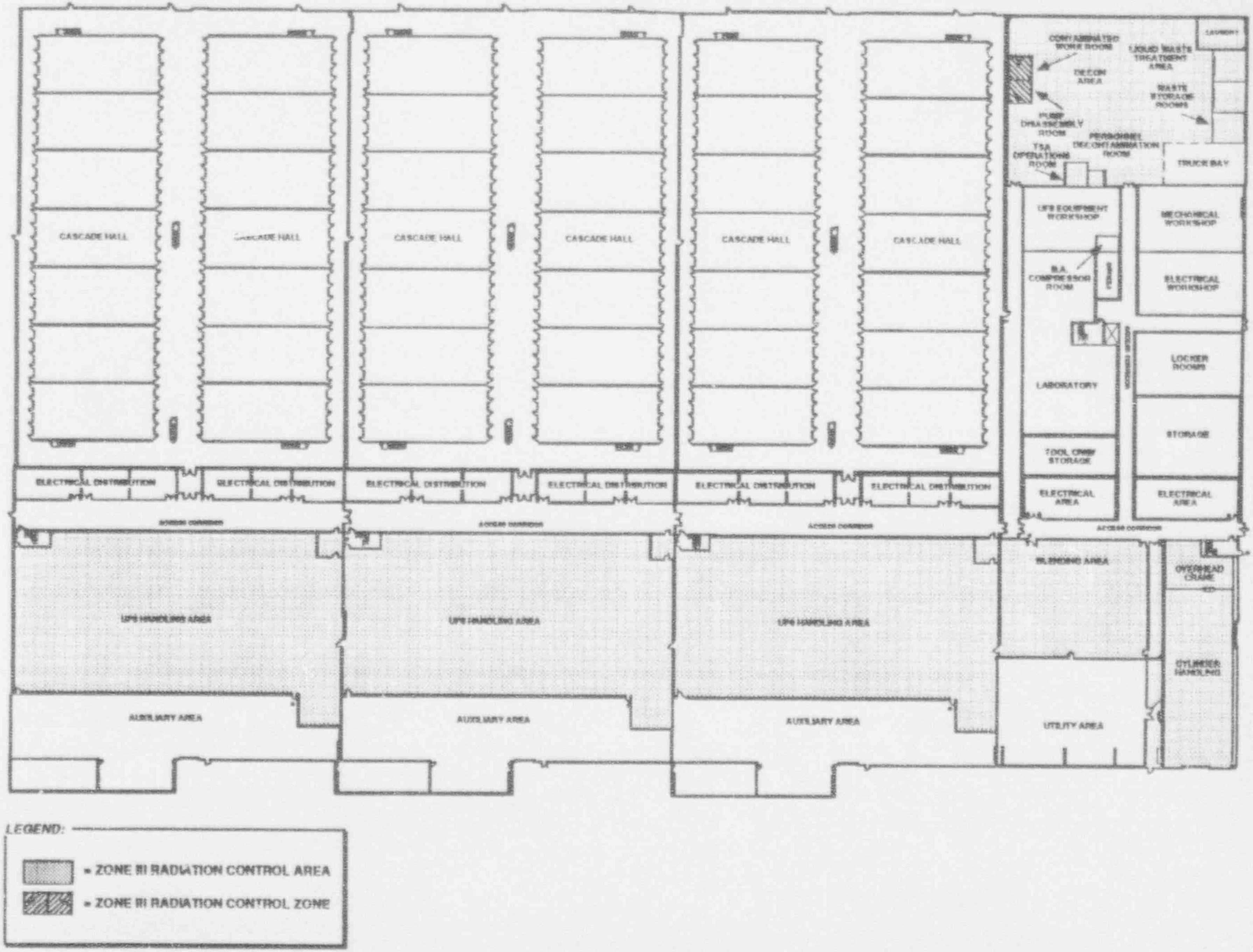

Figure 8.2 Claiborne Enrichment Center radiological access zones (LES, 1993a)

In combination, the NRC staff believes the thresholds specified in these definitions for an RCA and an RCZ will adequately identify areas where access control for the purpose of radiation protection is needed. Because of the nature of the operations to be conducted at the CEC, the first and second thresholds would not be reached except under highly unusual circumstances. The third and fourth thresholds would normally determine the RCAs and RCZs.

A large fraction of the collective worker dose is expected to be received during maintenance of equipment. LES is committed to designing processes and equipment that contain radioactive material to be as maintenance free as practicable. Additional reductions in occupational exposures may be achieved by other good practices such as:

- removing as much radioactive material as possible from equipment (for example, use of portable vacuum pump sets for purging UF_6 lines) and the surrounding area prior to maintenance, thereby lowering potential internal and external exposure

- providing adequate space for ease of maintenance thus reducing the length of time required to complete the task and the time of exposure

- preparing and using procedures that contain specifications for tools and equipment needed to complete the job

- proper job planning, including practice on mockups

- reviewing previous similar jobs

- identifying and communicating the highest contamination areas to the workers prior to the start of work.

Equipment which can be removed from UF_6 process lines and decontaminated prior to maintenance will be taken to the Decontamination Workshop, which is a specialized decontamination and maintenance area in the Separations Building. Equipment decontaminated in this specialized area will include items such as pumps, valves, flexible connectors, traps, sample manifolds, sample bottles, instruments, and piping sections. The Decontamination Workshop will also decontaminate other equipment used to process effluents in the Separations Building such as waste handling pumps, valves, tools, and miscellaneous piping sections and equipment. Decontamination of scrap metal can also be carried out to reduce the volume of radioactive solid waste from the site (SAR, Section 8.2.4).

LES has committed to using gloveboxes designed to maintain at least 0.1 inches of water differential pressure anytime that use of the glovebox is likely to result in exceeding the "Airborne Radioactivity Area" limits of 10 CFR 20.1003, and will cease using any glovebox until the required differential pressure has been restored (LES, 1993e). Since an exposure at the radiological limit as specified in 10 CFR 20.1003 for one week could result in an intake of soluble uranium in excess of 10 milligrams (mg), and a differential pressure of 0.1 inches of water as indicated by a gauge may not provide adequate air inleakage into the glovebox during normal operations and in the event of failure of the glovebox, the NRC staff recommends the following license condition:

> Notwithstanding the requirements related to gloveboxes in Section 3.2.5.1 of the applicant's Proposed License Conditions, gloveboxes shall be designed to maintain a negative differential pressure of 0.25 inches of water. This differential pressure shall be maintained anytime that the glovebox is in use. If the differential pressure is lost, use of the glovebox will cease until the required differential pressure is restored.

LES has also committed to maintaining air flow rates with a minimum linear face velocity of 100 feet per minute for hoods (during use) and air flow rates at other exhausted enclosures and close-capture points adequate to prevent release of airborne uranium to the work areas. In addition, LES has committed to checking air flow rates on a monthly basis and after modification of a hood, exhausted enclosure, close-capture point equipment, or the ventilation systems serving these barriers. The NRC staff finds these provisions acceptable.

LES is committed to providing monitoring stations at RCA boundaries, and a monitor (frisker), step-off pad and container for discarded protective clothing at each egress point from an RCZ (LES, 1993e). In addition, LES will post the most recent survey information regarding radiation, and contamination levels at each access point to an RCA or RCZ (LES, 1993a). This is acceptable to the NRC staff. Further, in accordance with 10 CFR 20.1902(e), LES must also post all areas where licensed material quantities are more than 10 times the values in Appendix C to 10 CFR 20.1001-20.2401.

LES is committed to retaining records in accordance with 10 CFR Part 20 and Section 2.9 of the Proposed License Conditions (LES, 1993e); see also the discussion in Section 10 of this document. Provisions will be made for easy retrievability of stored records, and storage of records such that they will be protected from fire, water, dust, extreme humidity and temperatures.

With the exceptions above, the NRC staff finds the general guidelines of the occupational radiation protection program proposed by LES to be consistent with good industry practice, the ANPR, and 10 CFR Part 20.

8.4.1 System of Exposure Controls and Exposure Experience

The new 10 CFR Part 20 annual limits for workers (NRC, 1991c) which CEC will be required to comply with are summarized below:

EXPOSURE CONDITIONS	NEW PART 20 EXPOSURE LIMITS
External Radiation	5 rem/y total effective dose equivalent (TEDE); includes summation of both external deep dose equivalent and internal committed effective dose equivalent (CEDE). Internal dose equivalents for each organ are multiplied by risk-based weighing factor and summed (except for lens of eye, skin and extremities). Lens of eye: 15 rem/y Hand, elbow, arm below elbow, foot, knee, and leg below knee: 50 rem/y shallow dose equivalent. Same limit for skin, with requirement for calculating maximum skin dose to 1 cm^2 area.
Internal Radioactivity	Annual Limit for Intake (ALI) based on exposure to 2,000 Derived Air Concentration (DAC) - hours per year. Organs are assigned weighing factors based on estimated risk/rem to that organ versus risk/rem for whole body irradiation, capping the dose limit at 50 rem/y to avoid non-stochastic effects. The inferred limits are: Gonads: 20 rem/y Breast: 33 rem/y Red Marrow: 42 rem/y Lung: 42 rem/y Thyroid: 50 rem/y Bone Surface: 50 rem/y Each of 5 highest remaining Organs: 50 rem/y Embryo/Fetus: 0.5 rem TEDE/y

To protect workers from chemical toxicity effects from inhalation of soluble (Class D) uranium, 10 CFR 20.1201(e) also limits worker intake to no more than 10 mg of soluble uranium in a week. In addition to meeting the NRC 10 CFR Part 20 requirements, the applicant has established an annual administrative radiation dose limit of 1.0 rem total effective dose equivalent (TEDE). This is acceptable to the NRC staff.

8.4.1.1 External Exposures

All personnel whose duties require them to enter the RCA will wear individual external radiation monitoring dosimeters (for example, thermoluminescent dosimeters, or TLDs). The dosimeters will be evaluated at least quarterly to evaluate external radiation exposures to assure workers do not exceed the 250 mrem per quarter action level. If the quarterly action level is exceeded, the applicant will determine the types of activities that contributed to the worker's external exposure, and document the investigation.

As shown below, LES has estimated the external radiation dose rates around UF_6 cylinders and in CEC work areas for normal operations, based on operating histories at similar facilities in Europe (LES, 1993a):

Location	Dose rate (mrem/hr)
Plant general area (excluding Separations Bldg.)	<0.01
Separations Building	0.05
Separations Bldg. (UF_6 Handling and Blending Areas)	0.1
Empty used UF_6 shipping cylinder	10 @ contact 1.0 @ 3 ft
Full UF_6 shipping cylinder	5.0 @ contact 0.2 @ 3 ft

1 mrem = 0.01 mSv

In addition, LES has estimated typical annual external doses expected for CEC operation (LES, 1993a), which are based on worker exposure histories at similar European facilities, as shown below:

Worker Classification	Annual Dose (mrem/yr)
General office staff	<5
General operations staff	20
Technical Services Area technicians	20
Maintenance technicians	30
UF_6 handlers	100

1 mrem = 0.01 mSv

The data presented in Table 8.3 demonstrate that for many years of operational experience in Europe with similar facilities, incorporating state-of-the-art designs and controls, very low occupational external exposures were experienced. Therefore, controlling the external quarterly doses below 250 mrem should be easily achieved at the CEC. For example, this

Table 8.3 Exposure histories for Urenco Facilities (LES, 1993l)

Almelo (UCN) Urenco Facility

Department	TLD Values (mrem/yr) Number of Monitored Workers	Group Average Dose	Group Minimum Dose	Group Maximum Dose
Enrichment (SP4) Operations (Total)	35 68	10	1	25
Blending Operations	6	40	25	60
Container Handling	5	100	60	250
Decontamination/ Decommissioning	7	2	1	30
Maintenance	4	4	1	10

Capenhurst (BNFL) Urenco Facility

Group	Mean Dose (mrem/yr) 1987	1988	1989
Hex Handling	93	75	103
Operations	16	13	15
Shift Engineering	27	18	26

Gronau (Uranit) Urenco Facility

Group	Average Exposure (mrem/yr)
Cylinder Handling Personnel	< 150
Process Operators	< 50
Supervisory Personnel	< 20

1 mrem = 0.01 mSv

experience indicates that the maximally exposed occupational groups are workers who handle containers or cylinders; (60 to 250 mrem per year at the Almelo Facility). Therefore, even with additional internal doses, the CEC should be able to maintain the TEDEs of workers within the annual 1 rem administrative limit. Given the low potential for high external radiation exposure in a gaseous centrifuge operation, the NRC staff finds the proposed external radiation monitoring program acceptable. The NRC staff concludes that the 1 rem administrative limit is, as proposed by LES, also a reasonable ALARA goal for the initial operation of the facility, and is therefore acceptable.

8.4.1.2 Internal Exposures

The applicant states in Section 8.2 of the SAR that internal exposures for CEC workers will be evaluated by bioassay procedures to determine intakes (LES, 1993a) . Bioassay procedures will include urinalysis, whole body counting, or equivalent. The applicant further states that continuous air monitoring in airborne radioactivity areas may be performed to complement the bioassay program. LES has committed to performing bioassays for all personnel for whom airborne radioactivity monitoring indicates an intake of 1 mg of soluble uranium, or more, in a week. Therefore, the bioassay program should be able to detect activity corresponding to a 1 mg in a week intake of Class D uranium (10 percent of the 10 mg in a week regulatory limit). Follow-up bioassay measurements will be conducted to determine the committed effective dose equivalent (CEDE) (LES, 1993e). It should be noted that, monitoring for internal doses (for Class D intakes), as required by 10 CFR 20.1502(b), is generally conducted in accordance with Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses" (NRC, 1992a). In addition, to demonstrate compliance with 10 CFR 20.1204, application of Regulatory Guide 8.9 "Acceptable Concepts, Models Equations, and Assumptions for a Bioassay Program," Revision 1 (July, 1993), and NUREG/CR-4884, "Interpretation of Bioassay Measurements" (Lessard et al, 1987) are useful.

LES proposes to evaluate internal radiological doses annually (LES, 1993e). This is acceptable to the NRC staff for routine operations where there is no evidence of a significant intake. The NRC staff concludes that meeting the 10 mg weekly intake limit for natural and depleted uranium (Class D) compounds over the entire year provides assurance that the Annual Limit on Intake (ALI) will not be exceeded. Inhalation of 10 mg of depleted uranium with an Activity Median Aerodynamic Diameter (AMAD) of 1.0 μm would result in a CEDE of less than 10 mrem.

LES proposes to use air monitoring data to determine when bioassays should be conducted, which is consistent with Regulatory Guide 8.34 guidance. LES has committed to performing bioassays on all personnel who are likely to have had an intake of 1 mg of soluble uranium (LES, 1993e). As proposed, LES would conduct a bioassay within 72 hours after a suspected or known exposure (based on daily or weekly air samples), and be able to detect 5 μg/l of Class D uranium in a 24-hour (1.4 liter) urine sample up to 10 days after an intake. If attempts to obtain a 1.4 liter, 24-hour urine sample fail within 10 days after an intake, LES proposes that the worker's intake will be estimated using "other data" which could include, for example, quantitative air measurements of the affected work area (LES, 1993e). This is acceptable to the NRC staff.

Using ICRP Report 54, "Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation," which utilizes the current ICRP metabolic models, the NRC staff estimates that an LLD of 5 μg/l for uranium in a 24-hour urine sample (1.4 liter) collected during the seventh day following intake (expected to contain about 1 percent of the initial intake) and subjected to the bioassay procedure proposed by LES (LES, 1993a,e) corresponds to an intake of about a 0.7 mg of soluble uranium. This is below the 1 mg weekly intake action level proposed by LES, and well below the 10 mg weekly intake limit. Even in the event a full 24-hour urine sample could not be collected, it would appear that less than half a 24-hour sample should be able to detect an intake of 10 mg or less. It should be noted that based on the metabolic model for uranium presented in ICRP-26 and ICRP-30, which provides data for an AMAD of 1.0 μm, the NRC staff calculated an approximate 30 percent increase in uranium deposition in the kidney for a 0.2 μm AMAD particle size. During normal CEC operations, the NRC staff expects any uncontained uranium in particulate form to have an AMAD greater than 0.3 μm. In addition, an evaluation of 31 workers accidentally exposed to natural uranium in 1986 indicates that the ICRP guidance may overestimate the amount of uranium present in urine at 7 days, since bioassay data indicates more rapid excretion of Class D uranium than originally believed (Fisher et al, 1990). This implies that the potential intake which might go undetected under the proposed bioassay program could be somewhat higher than estimated using the ICRP model. Nevertheless, the NRC staff concludes that it is unlikely that these potential nonconservatisms would account for as much as a factor of 10, and an intake involving more than 10 mg of soluble uranium would go undetected for an entire week. In addition, the assurance that LES should normally be able to measure a 1 mg intake of soluble uranium is acceptable to the NRC staff.

LES has committed to restricting workers from activities that could routinely or accidentally result in internal exposures to soluble uranium until a urine analysis result is less than a threshold value of 15 μg/l (LES, 1993e). This is acceptable to the NRC staff, since a uranium concentration of 15 μg/l in a 24-hour urine sample will most likely ensure that the uranium kidney burden is low enough so as not to provide a significant cumulative toxic effect to the kidney. For example, according to NUREG/CR-4884, approximately one month after an intake of 10 mg of soluble uranium, a 24-hour urine sample would contain uranium at a concentration of approximately 15 μg/l. The kidney burden of uranium would decline from less than 2.5 mg in the first 24-hour period after exposure to less than 0.5 mg after one month.

8.4.1.3 Monitoring for Airborne Radioactivity in the Workplace

LES has committed to providing alpha-in-air monitors (continuous air monitors) to be used in areas controlled for radiation which provide active (on-line) monitoring for gross alpha with an LLD of 3.7×10^{-3} Bq/m^3 (1×10^{-13} μCi/ml), or 0.02 mg uranium in a total sample (LES, 1993e). These proposed LLDs are acceptable to the NRC staff. The radiological LLD would be about 0.02 percent of the values listed in Appendix B of 10 CFR 20.1001-20.2402, Table 1, Column 3 for Class D uranium (LES, 1993e). For a U-238 average air concentration of 1×10^{-13} μCi/ml, the weekly intake via inhalation by occupational workers would be about 0.014 mg of uranium, assuming a breathing rate of 1.2 m^3/hr and a 40-hour occupancy time. Typical air flow rates for general area air samplers and lapel air samplers are 1.2 to 1.8 m^3/hr and 0.12 to 0.30 m^3/hr, respectively. A general area air filter sample collected after a single shift (8 hours), having 0.02 mg of U-238, would indicate an average air concentration of about 0.002 mg/m^3, which is approximately one percent of the toxicological DAC of 0.2 mg/m^3. Similarly, a lapel air sampler with 0.02 mg of U-238 would indicate an average air concentration of about 0.02 mg/m^3 (approximately 10 percent of the toxicological DAC). This assessment assumes that a workers lungs would filter uranium in a manner similar to filters installed in air samplers.

LES will take action when airborne concentrations might result in an intake of 1 mg of soluble uranium (10 percent of the 10 mg in a week limit; LES, 1993e). Permanently mounted air monitors will be located to provide representative samples of the work station air of workers (LES, 1993e). The proposed types and locations of 44 airborne monitors are shown in Table 8.4. As shown, radioactivity monitors that have alarm capability are not integrating monitors. In addition, several portable monitors are identified for use where necessary, such as in temporary RCZs (LES, 1993a). Alarming air monitors used in airborne radioactivity areas will alert workers to the presence of airborne radioactivity in their work areas.

The monitors which have alarm capability will be calibrated and set such that when radioactivity concentrations corresponding to 3 ppm (2.4 mg/m^3) of HF, or greater, are detected, the units will alarm. The NRC staff is in agreement with the monitor alarm setpoints of 3 ppm of HF proposed by the applicant. Assuming that all alpha particles originate from U-238 and the entire quantity of released UF_6 reacts with water, the NRC staff calculated a U-238 air concentration equivalent to 3 ppm HF at standard temperature and pressure to be about 10 Bq/m^3 (0.25 pCi/l), which is above the air concentration LLD of 0.004 Bq/m^3 (0.0001 pCi/l) proposed by the applicant for these monitors. A uranium concentration of 10 Bq/m^3 would result in a worker's uranium intake rate of less than 1 mg/hr which, for soluble uranium, corresponds to a radiological dose rate of less than 10 µSv/hr (1 mrem/hr). It should be noted that the 3 ppm HF level is approximately half the American Industrial Hygiene Association's Emergency Response Planning Guideline Level 1 (ERPG-1). ERPG-1 (4.1 mg/m^3 HF) is a level that nearly all individuals could be exposed to for up to an hour without experiencing other than mild, transient effects or without experiencing an objectionable odor. Alarms will be sounded locally and in the Central Control Room. LES proposes to shut down ventilation systems to any affected areas upon alarm activation (LES, 1993g).

During the preoperational inspection, NRC staff will confirm that monitor locations and their alarm set-points are adequate.

Continuous monitor filters representing integrated air samples during the sampling period will be collected weekly or following any indication of a release of radioactivity to a work area that is likely to result in a soluble uranium intake in excess of 1 mg, or 10 percent of the 10 CFR 20.1201(e) limit (LES, 1993e). Investigations would be performed if airborne

Table 8.4 Continuous air monitors, capabilities and locations

MONITOR NUMBER	TYPE	ALARM	INTEGRATING	LOCATION
1	α-in-air	No	Yes	Radioactive Waste Storage Area Ventilation Return
2	HF	Yes	No	Pump Disassembly Room Vestibule
3	α-in-air	Yes	No	Pump Disassembly Room Vestibule
4	α-in-air	a	No	Contaminated Equipment Workroom Ventilation Return
5	α-in-air	a	No	Pump Disassembly Room Ventilation Return
6	α-in-air	a	No	Pump Disassembly Room Ventilation Return
7	α-in-air	a	No	Pump Disassembly Room Ventilation Return
8	α-in-air	No	Yes	Contaminated Equipment Workshop Ventilation Return
9	α-in-air	No	Yes	Technical Services Area Corridor
10	α-in-air	Yes	No	Decontamination Bath Monorail A
11	α-in-air	Yes	No	Decontamination Bath Monorail B
12	α-in-air	No	Yes	Decontamination Workshop Ventilation Return
13	α-in-air	No	Yes	Truck Bay Ventilation Return
14	α-in-air	Yes	No	Tank Room Sample Sink
15	α-in-air	No	Yes	Effluent Collection Pit/Tank Room Ventilation Return
16	α-in-air	No	Yes	Laundry Room Ventilation Return
17	HF	Yes	No	UF_6 Sample Room Ventilation Return
18	HF	Yes	No	Sample Storage Room Ventilation Return
19	α-in-air	No	Yes	Health Physics Laboratory Ventilation Return
20	α-in-air	No	Yes	Chemical Laboratory Ventilation Return
21	α-in-air	No	Yes	Plant Entrance Corridor (used as a "control")
22	α-in-air	Yes	No	Unit 1 UF_6 Handling Area[b]
23	α-in-air	No	Yes	Unit 1 UF_6 Handling Area Ventilation Return
24	α-in-air	Yes	No	Unit 2 UF_6 Handling Area[b]
25	α-in-air	No	Yes	Unit 2 UF_6 Handling Area Ventilation Return
26	α-in-air	Yes	No	Unit 3 UF_6 Handling Area[b]
27	α-in-air	No	Yes	Unit 3 UF_6 Handling Area Ventilation Return

Table 8.4 Continuous air monitors, capabilities and locations (continued)

MONITOR NUMBER	TYPE	ALARM	INTEGRATING	LOCATION
28	α-in-air	Yes	No	Blending Area[b]
29	α-in-air	No	Yes	Utility Area Ventilation Return (used as a "control")
30	α-in-air	Yes	No	Portable (Technical Services Area)
31	α-in-air	Yes	No	Portable (Technical Services Area)
32	α-in-air	Yes	No	Portable (Technical Services Area)
33	α-in-air	Yes	No	Portable (Technical Services Area)
34	α-in-air	Yes	No	Portable (Technical Services Area)
35	α-in-air	Yes	No	Portable (Technical Services Area)
36	α-in-air	Yes	No	Portable (Unit 1)
37	α-in-air	Yes	No	Portable (Unit 1)
38	α-in-air	Yes	No	Portable (Unit 1)
39	α-in-air	Yes	No	Portable (Unit 2)
40	α-in-air	Yes	No	Portable (Unit 2)
41	α-in-air	Yes	No	Portable (Unit 2)
42	α-in-air	Yes	No	Portable (Unit 3)
43	α-in-air	Yes	No	Portable (Unit 3)
44	α-in-air	Yes	No	Portable (Unit 3)

[a] Monitor has indicating display in place of an alarm.
[b] The exact location of this monitor will be specified prior to the NRC preoperational inspection.

radioactivity were to exceed the action levels. Corrective action would include investigation and evaluation of the need for changes, consistent with ALARA principles (LES, 1993e). In addition, filters will be collected each shift following change in process equipment or control, and following detection of any event (such as leakage, spillage, or blockage of process equipment) that might result in the intake considered above.

LES is committed to checking the representativeness of the work station air samplers annually, and when significant process or equipment changes have been made (LES, 1993e). As noted below, the NRC staff will also confirm the representativeness of work station air samplers during the preoperational inspection. LES also proposes to substitute continuous air samplers or personnel lapel air samplers with periodic sampling in plant areas where conditions favor periodic sampling (for example, areas within UF_6 processing areas, decontamination areas, waste processing areas, and laboratories) where normal continuous monitoring may not be reasonably achieved (LES, 1993e).

LES is committed to designing ventilation equipment such that normal air flow or leakage flows are generally from areas of lesser contamination to areas of higher potential contamination. During the preoperational inspection, NRC staff will confirm that air flow is consistent with the design air flow discussed above, and that the air samplers are properly placed for prompt detection of releases and for representative sampling of the work areas. If air samplers cannot be placed to perform both functions, LES will have the latitude to consider use of lapel air samplers or other methods to achieve representative measurements of worker breathing zones.

The NRC staff finds the proposed airborne radioactivity monitoring program acceptable for occupational protection and in accordance with 10 CFR Part 20 requirements.

8.4.1.4 Control of Surface and Personnel Contamination

As discussed in Section 8.2, LES will provide various methods for the control and containment of uranium in all areas of the facility to minimize worker exposures. In contaminated areas, LES proposes that worker exposure to surface contamination will be minimized by proper use of protective clothing and equipment. LES is committed to providing protective clothing (appropriate for the existing radiological conditions), which will be worn in RCZs. During NRC's preoperational inspection, the NRC staff will review LES' procedures that specify requirements for protective clothing for each RCA and RCZ area in the CEC.

LES has committed to limiting skin or personal clothing contamination at egress from RCAs and RCZs to no more than 150 dpm/100 cm^2 of alpha, or beta/gamma (LES, 1993e). LES' action levels for surface contamination (alpha, or beta/gamma) on laundered protective clothing are 150 dpm/100 cm^2 for RCAs and 1,000 dpm/100 cm^2 for RCZs (LES, 1993e). These action levels are acceptable to the NRC staff. It should be noted that most of the radioactive material that could contaminate protective clothing at the CEC will be in soluble form and therefore will be readily removed by laundering.

LES also proposes that if areas containing removable surface contamination can be isolated from uncontaminated adjacent work areas using a barrier such that contamination cannot be dispersed to these areas, personnel working in the "clean" adjacent areas need not wear protective clothing (LES, 1993e). The NRC staff finds this proposal acceptable.

LES has committed to providing routine contamination survey monitoring in all UF_6 process areas, with routine, periodic checks of non-UF_6 process areas including those areas that are normally free of contamination. Moreover, LES has committed to surveying RCAs and RCZs at least weekly, and to surveying lunch rooms and change rooms at least daily (LES, 1993e). This is acceptable to the NRC staff. Monitoring will include measurements of fixed and removable surface contamination, with extent and frequencies based on the potential for contamination in each area, and operational experience. Removable surface contamination will be considered to be uranium that can be transferred to a dry smear paper with moderate pressure. Survey instruments and methods would be capable of detecting alpha contamination at and below the levels discussed above, using proportional counters, alpha scintillation counters, thin-window GM counters and other instruments as appropriate.

LES has defined a contaminated area as an area where removable contamination levels are above 20 dpm/100 cm^2 alpha or 1,000 dpm/100 cm^2 beta/gamma (LES, 1993a). LES has committed to RCZ cleanup action levels of 5,000 dpm/100 cm^2 (alpha or beta/gamma) for removable surface contamination, and 250,000 dpm/100 cm^2 (alpha, or beta/gamma) for fixed surface contamination (LES, 1993e). LES has also committed to initiating cleanup of RCZs within 24-hours after detection of removable surface contamination exceeding 5,000 dpm/100 cm^2 (alpha, or beta/gamma; [LES, 1993e]). The RCZ cleanup action level proposed by LES is acceptable to the NRC staff. This conclusion is based on the following results of NRC staff evaluations. For removable surface contamination (U-238) of 5,000 dpm/100 cm^2 averaged over an entire facility, using a resuspension factor of $5x10^{-5}$ per meter (IAEA, 1970), the NRC staff calculated a weekly intake (40-hour exposure) via inhalation of less than 2 mg of uranium. For fixed uranium surface contamination of 250,000 dpm/100 cm^2 applicable to the CEC (5 percent enrichment), and assuming an infinite planar source and 100 percent occupancy, the NRC staff calculated an annual deep dose equivalent of less than 40 mrem. For transfer of material and equipment to unrestricted areas and release from the facility for unrestricted use, the applicant has committed to meeting surface contamination guidelines prescribed in "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of License for Byproduct, Source, or Special Nuclear Material" (NRC, 1993c). Because current regulations do not contain specific criteria by radionuclide for unrestricted release of material and equipment, the NRC staff finds the applicant's proposal acceptable.

8.4.1.5 Respiratory Protection Program

LES must utilize any respiratory protection in accordance with 10 CFR Part 20, Subpart H, which provides the requirements for an acceptable respiratory protection program. As stated, respiratory protection is only to be relied on when process or other engineering controls are impracticable. Additional guidance is contained in Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection" (NRC, 1976).

During the preoperational inspection the NRC staff will review the applicant's written procedures for respiratory protection and confirm that the respiratory protection program is consistent with 10 CFR Part 20, Subpart H.

8.4.1.6 Instrumentation, Calibration, and Maintenance Program

LES proposes to use two basic types of personnel monitoring instruments at CEC; friskers and hand and foot monitors (LES, 1993a,e). In the preoperational inspection, NRC staff will confirm that the frisker and hand and foot monitors are capable of measuring low-level alpha and beta activity on protective clothing and equipment and surfaces, and direct radiation, at the CEC.

Section 20.1501(b) of Title 10 of the Code of Federal Regulations requires licensees to provide assurance that instruments and equipment used for quantitative radiation measurements (dose rate and effluent monitoring) are calibrated periodically for the radiation measured. The National Council on Radiation Protection and Measurement states that the required frequency of calibration ranges from once every few weeks to annually depending on the amount of use an instrument receives, the environmental condition it is used under, and the historical experience of each instrument type (NCRP, 1991).

NRC licensees must make routine survey measurements with reasonable accuracy and reliability. Reliability is a function of the detector systems, instrument usage, manufacturing quality, and the user's calibration and maintenance programs. LES has proposed to calibrate instruments prior to initial use and to subsequently perform annual calibrations or calibration verifications (LES, 1993e). The NRC staff finds that verification of an instrument's calibration as proposed by LES may not sufficiently ensure instrument accuracy over its entire range of measurement and that more frequent calibrations may be recommended by the instrument manufacturer. In addition, the NRC staff recommends that LES perform periodic checks of the proper operation of radiation detection and measurement systems, and conduct proper maintenance to assure their continued reliable operation, consistent with the manufacturer's recommendations. Therefore, the NRC staff recommends the following license condition:

> Notwithstanding the instrument calibration requirements in Section 3.2.4 of the applicant's Proposed License Conditions, instruments used for radiation protection purposes shall be calibrated before initial use and undergo periodic operability checks in accordance with written established procedures. If an instrument fails an operability check or has undergone repair or any modification that could affect its proper response, it shall be recalibrated. Instruments shall be recalibrated at least annually or according to the manufacturer's recommendations, whichever is more frequent.

LES has committed to using calibration sources traceable to the National Institute of Standards (NIST; formerly the Bureau of Standards), or equivalent, and which are ± 5 percent of the stated values (LES, 1993e). In addition, LES has committed to determining, on a daily basis (less frequently only if required by long counting intervals), the background and efficiency of laboratory counting instruments used for radiation protection purposes (LES, 1993e). These commitments are acceptable to the NRC staff.

Section 20.1204 of Title 10 of the Code of Federal Regulations allows estimates of worker inhalation intakes of radioactive materials based on air sampling. LES has proposed to calibrate air flow measurement devices prior to initial use, and to subsequently perform an annual calibration or calibration verification (LES, 1993e). The NRC staff finds that calibration verification of an air flow meter as proposed by LES may not adequately ensure the device's accuracy over its entire range of measurement, and recommends the following license condition:

> Notwithstanding the calibration requirements for air flow measurement devices in Section 3.2.4 of the applicant's Proposed License Conditions, flow rate meters or devices used to measure flow rates for air or effluent sampling shall be calibrated in accordance with procedures at least annually and after modifications or repairs to the meter, and when the meter is believed to have been damaged.

This is in accordance with the guidance in Regulatory Guide 8.25, "Air Sampling in the Workplace" (NRC, 1992b). NRC staff will confirm the adequacy of procedures related to instrument calibration during the preoperational inspection and during periodic operational inspections.

8.4.1.7 Radiation Work Permit System

LES is committed to establishing a Radiation Work Permit (RWP) system that will result in posting of RWPs in an "information area" in all locker rooms where the RWPs can be read prior to entering RCZs within the facility. RWPs will provide personnel with information relating to the radiation levels and contamination control procedures for various activities (LES, 1993a and LES,1993e).

LES has also committed to issuing RWPs for activities involving licensed materials not covered by operating procedures, and where radioactivity levels are likely to exceed airborne radioactivity limits specified in 10 CFR 20.1003, or wherever deemed necessary by the HP Manager to maintain doses ALARA. Criteria for ensuring that RWPs are issued and closed out properly are proposed to be as follows (LES, 1993e):

- The HP Manager or designee is responsible for determining the need for, issuing, and closing out RWPs

- Planned activities or changes to activities inside RCAs and RCZs or with licensed materials shall be reviewed by the HP Manager or designee for potential for causing radiation exposure to exceed action levels and radioactive contamination

- RWPs shall include requirements for any necessary safety controls, personnel monitoring devices, protective clothing, respiratory protection equipment, air sampling equipment, and health physics coverage needed for the activity

- Copies of current RWPs shall be posted at the location of the work area
- RWPs shall clearly define and limit the work activities to which they apply. The RWPs shall be closed out when the applicable work activities are terminated.

The NRC staff concludes that the RWP system proposed by the applicant is acceptable.

8.4.2 Conclusion

With the exceptions noted, the NRC staff finds the occupational radiation protection program proposed by LES to be consistent with good industry practice, the ANPR, and 10 CFR Part 20, and acceptable.

8.5 Public Radiation Exposure

Public exposure to uranium may result from small, controlled releases from the uranium enrichment process lines, during decontamination and maintenance of equipment, from releases of radioactive liquids to surface water, and from transportation of UF_6 cylinders. Direct radiation (sky shine) in offsite areas is expected to be undetectable since the photons associated with the uranium will be almost completely absorbed by the heavy process lines, equipment, and tanks to be employed at CEC.

Part 20 of Title 10 of the Code of Federal Regulations provides an explicit TEDE dose limit for the public of 0.1 rem/yr from all sources, and includes both internal and external doses through all pathways (including food). In an uncontrolled area, external dose rates cannot exceed 2 mrem in any one hour. Concentration limits for radioactivity in air and water provided in 10 CFR Part 20, Appendix B, Table 2 must be complied with at the CEC. Further, LES will be subject to EPA's generally applicable standards in 40 CFR Parts 61 and 190.

The principal source of public exposure, although small, is expected to be from atmospheric releases as the facility design is currently proposed. Such releases would be primarily controlled through the Separations Building ventilation system (LES, 1993a, p. 8.3-1). All air to be released from potentially contaminated areas of the facility would be filtered by prefilters and high efficiency particulate air (HEPA) filters to remove most of any particulate radioactivity in effluents prior to discharge. LES has committed to testing newly installed GEVS and TSA HEPA filter systems for particulate removal efficiency, and to measuring the differential pressure across these HEPA filters monthly, or to automatically monitoring and alarming the differential pressure (operating procedures will specify the limits/setpoints according to the manufacture's recommendations (LES, 1993e). The NRC staff finds this acceptable.

LES has committed to implementing a radiological environmental monitoring program prior to plant operation (preoperational) and after CEC start-up (LES, 1993a and LES, 1993e). The preoperational environmental monitoring program for the site and surrounding environs will be established before the CEC begins operation to provide background data on preexisting radiation levels and to provide information for critical pathway analysis (see Section 8.5.2). As new data become available after start-up, the environmental program may be revised to provide more useful information (for example, changing sample types, locations, etc). The background information will permit comparisons with radioactivity in biota, air, and water around CEC after operations commence to be certain that there is no unexpected buildup of radioactivity in the environment as a result of plant operations. The monitoring program will also support the radiological compliance program, since it will provide assurance that the process and effluent control systems are operating properly. Environmental measurements will also assist with estimates of potential radiological impacts on local residents in the event detectable radioactivity is found from normal operations or accidents. The results of the operational environmental monitoring program will be submitted biennially to the NRC for review. The environmental monitoring program is described in detail in the NRC Draft Environmental Impact Statement (DEIS) on the CEC (NRC, 1993a).

8.5.1 Dose Evaluation Methods

Radioactive material released to the atmosphere and surface water is dispersed during transport through the environment and transferred to human receptors through inhalation, ingestion, and direct exposure pathways. Therefore evaluation of impacts requires consideration of potential receptors, environmental transport, exposure pathways and conversion of estimates of intake to dose. This section presents a discussion of the approach used in this SER.

This SER assessment of radiological impact considers the entire population surrounding the proposed CEC within a distance of 80 kilometers (50 miles) and those individuals whose exposure would bound all foreseeable impacts related to CEC operation. The total population considered numbers 349,000 and the distribution by area is presented in Table 2.1 of this SER. The three individuals whose exposure would bound potential impacts were assumed to be located 800 meters north of the plant stacks at a permanent residence, 570 meters south-southeast of the plant stacks at the edge of Bluegill Pond, and 6,500 meters south of the plant stacks at the northern edge of Lake Claiborne. The atmospheric dispersion modeling discussed in Chapter 4 predicted that the maximum annual average air concentration of radioactive material would occur approximately 800 meters north of the plant stacks. Therefore the individual assumed to be located 800 meters north of the plant stacks is the maximally exposed individual for atmospheric releases. Annual average air concentrations for the Bluegill Pond and nearest resident (475 m north of the plant stacks) locations are approximately 20 percent less than the maximum values. As a consequence of adoption of conservative assumptions for drinking water and irrigated food consumption, the Bluegill Pond resident is the maximally exposed individual for the normal operational impact analysis.

The primary component of atmospheric dispersion is mechanical mixing produced by temperature and wind velocity gradients. For projected normal operational releases the methods of Regulatory Guide 1.111 (NRC, 1977a) are used to estimate concentrations of released material at a range of distances and directions from the release point. These methods use the Gaussian plume dispersion model and are implemented in the XOQDOQ computer code (Sagendorf et al, 1982). Concentrations per unit release quantity (that is, χ/Q values) predicted using this model and appropriate meteorological data are summarized in Table 2.8 of this SER. The primary component of dispersion during liquid transport is dilution due to mixing of stream and river flows. A simple material balance model using site specific CEC ER and SAR surface water hydrology data is used to estimate the degree of dilution and related concentrations of released material throughout the environment (LES, 1993a).

Members of the public may be exposed to radioactive material dispersed in the environment through inhalation of air, ingestion of drinking water, ingestion of terrestrial foods and animal products, inadvertent ingestion of soil, and direct irradiation from nuclides deposited on the ground or present in surface water. Guidance on acceptable exposure models for these pathways has been published in NRC Regulatory Guide 1.109 (NRC, 1977b) and incorporated into a variety of computer codes. The GENII code (Napier et al, 1988) is used to estimate doses in this SER. To the extent possible, modeling parameters, such as age specific inhalation and food consumption rates, were those recommended in Regulatory Guide 1.109 (NRC, 1977). For the purposes of these evaluations, individuals were assumed to derive their entire terrestrial food and animal product food consumption from locally grown contaminated crops.

Radionuclide uptake rates estimated with the environmental transport and exposure pathway models were converted to dose equivalent using metabolic and physical distribution and energy deposition models. For the evaluations of this SER the dose conversion approach and models recommended in ICRP-26 and -30 (ICRP, 1977 and 1979) were used. Dose conversion factors (DCF's) for adults were those published in NUREG/CR-0150 (NRC, 1981b) and doses estimated for this age category were converted to dose estimates for the teen, child, and infant age categories using the relative age-specific dose factors published in NUREG/CR-4628 (NRC, 1986a). Tissue specific dose conversion factors used for comparison with the 40 CFR Part 190 criteria (EPA, 1977) were those published in NUREG/CR-0150. The DCF's provide an estimate of the committed effective dose equivalent (CEDE) that would be incurred over a fifty year period due to one year of exposure. In all cases, the released nuclides were assumed to be a soluble form of the uranium-234 isotope as recommended in ICRP-26 for UF_6 and related compounds. Since the released particles would be formed from vapor phase condensation and would be filtered through HEPA filters, the average particle diameter at the point of release would be small. In order to provide a conservative impact analysis the particle diameter in the gaseous effluents was assumed to be 0.3 microns, and the dose conversion factors were modified according to NUREG/CR-0150.

8.5.2 Dose Estimates For Atmospheric Releases

The CEC releases radioactive material to the atmosphere through stacks 36.6 meters (120 ft.) tall. The estimated source term evaluated in Section 7.4 of this SER could be as high as 4.4 MBq per year (120 microcuries per year) of uranium isotopes. Expected exposure pathways include inhalation of air and direct exposure from material deposited on the ground. In addition to these expected routes of exposure, members of the public may also consume food contaminated by deposited radionuclides and inadvertently ingest contaminated soil resuspended from the ground. Potential tissue and effective doses for the maximally exposed adult individuals and the population are presented in Table 8.5.

Table 8.5 Potential doses to adult individuals and the population from atmospheric releases

Tissue	Maximally Exposed Individual Doses (Sv)[a]			
	Bluegill Pond	Lake Claiborne	800 Meter Resident	Population (person-Sv)[b]
Gonads	$8.8x10^{-12}$	$2.8x10^{-12}$	$3.0x10^{-11}$	$1.1x10^{-7}$
Breast	$5.6x10^{-12}$	$1.8x10^{-12}$	$2.0x10^{-11}$	$7.2x10^{-8}$
Red Bone Marrow	$2.3x10^{-10}$	$6.8x10^{-11}$	$7.6x10^{-10}$	$2.7x10^{-6}$
Lung	$8.4x10^{-11}$	$2.6x10^{-11}$	$2.8x10^{-10}$	$1.2x10^{-6}$
Thyroid	$8.8x10^{-12}$	$2.8x10^{-12}$	$3.0x10^{-11}$	$1.1x10^{-7}$
Bone Surface	$3.5x10^{-9}$	$1.0x10^{-9}$	$1.2x10^{-8}$	$3.9x10^{-5}$
Stomach	$1.0x10^{-12}$	$3.2x10^{-13}$	$3.5x10^{-12}$	$1.2x10^{-8}$
Small Intestine	$1.4x10^{-12}$	$4.4x10^{-13}$	$4.8x10^{-12}$	$1.5x10^{-8}$
Upper Large Intestine	$4.8x10^{-12}$	$1.4x10^{-12}$	$1.6x10^{-11}$	$4.4x10^{-8}$
Lower Large Intestine	$1.6x10^{-11}$	$4.8x10^{-12}$	$5.2x10^{-11}$	$1.4x10^{-7}$
Kidney	$1.7x10^{-9}$	$5.2x10^{-10}$	$5.6x10^{-9}$	$2.0x10^{-5}$
CEDE	$2.5x10^{-10}$	$7.6x10^{-11}$	$8.0x10^{-10}$	$2.8x10^{-6}$

[a] Sieverts (Sv) = 0.01 x rem

[b] Person-Sv = 0.01 x Person-rem

The inhalation and food ingestion pathways each contribute approximately half of the total dose. Dose contributions from the external exposure pathways are approximately one-millionth of the total projected dose. Potential doses estimated for maximally exposed individuals in the teen, child, and infant age categories are somewhat higher than the adult doses presented in Table 8.5. An infant located at the 800 meter-location would be the critical individual for the air pathway and could receive a CEDE of 2.4×10^{-9} Sv (2.4×10^{-7} rem). The largest tissue dose would be 5.4×10^{-8} Sv (5.4×10^{-6} rem) to the bone surface of the infant. For both maximally exposed individuals and members of the population the estimated doses are a small fraction of the dose that the individual would receive from natural background sources.

8.5.3 Dose Estimates For Liquid Releases

Radioactive material would be released from the proposed CEC to surface water in Bluegill Pond as a consequence of normal operations. The low levels of radionuclides would travel with the water from the pond west to Cypress Creek and southward through Lake Claiborne towards the Gulf of Mexico. The estimated source term evaluated in Section 7.4 of this SER could be as high as 1.0 MBq per year (28 microcuries per year) of uranium isotopes.

Potential tissue and effective doses for the maximally exposed adult individuals and the population are presented in Table 8.6. Dose estimates are not developed for the northern location (that is, 800-meter resident) as this location does not have access to potentially contaminated water. Potential exposure pathways include drinking water ingestion, terrestrial and animal product food ingestion, fish and seafood ingestion, and direct exposure during recreational activities (that is, fishing, swimming, and boating). The drinking water ingestion, food ingestion, and fish ingestion pathways each would contribute approximately one-third of the total dose. Potential doses from the direct exposure pathways would be approximately one-millionth of the total estimated dose. Potential doses estimated for the maximally exposed individuals in the teen, child, and infant age categories range from two to ten times the doses presented in Table 8.6. The critical individual for the liquid pathway would be an infant located at Bluegill Pond. The potential CEDE estimated for this infant would be 6.0×10^{-6} Sv (6.0×10^{-4} rem) and the largest tissue dose would be 1.4×10^{-4} Sv (1.4×10^{-2} rem) for the bone surface. For the maximally exposed individual and members of the population the estimated doses are a small fraction of the dose from background radiation sources.

8.5.4 Evaluation of Cumulative Radiological Impact from Routine Operations

NRC regulations (10 CFR Part 20, Subpart D) require that the total effective dose equivalent (TEDE) for members of the public for routine operations not exceed 100 mrem in a year. In addition, EPA regulations, 40 CFR Part 190 and 40 CFR Part 61, address emissions to the general environment and to the atmosphere. For routine releases to the general environment, 40 CFR Part 190 requires that the annual dose equivalent not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ (EPA, 1977). For releases to the atmosphere, 40 CFR Part 61 requires that the annual effective dose equivalent not exceed 10 mrem (EPA, 1973). The maximum potential cumulative annual impact to any individual for potential CEC atmospheric and liquid releases was estimated to be a CEDE of 6.0×10^{-6} Sv (0.6 mrem) to an infant located at Bluegill Pond. The maximum organ dose was estimated to be 3.7×10^{-5} Sv (3.7 mrem) to the whole bone. For atmospheric releases the maximum CEDE was estimated as 2.4×10^{-9} Sv (2.4×10^{-4} mrem) for the infant located 800 meters north of the plant stacks. It is noted that these dose estimates assume that the infant's milk is produced by milk cows whose entire liquid intake comes from Bluegill Pond water. Even under the conservative assumptions used in the analysis, the maximum doses are well within limits set by the NRC and EPA.

The NRC staff concludes from this analysis that the proposed combination of design and administrative controls will permit LES to operate the CEC facility such that any individual outside the site boundary will be exposed to radiation doses (TEDE) that are no more than a small fraction of the NRC limits specified in 10 CFR Part 20, Subpart D.

Table 8.6 Potential doses to adult individuals and the population from liquid releases

Tissue	Maximally Exposed Individual Doses (Sv)[a]		
	Bluegill Pond	Lake Claiborne	Population (person-Sv)
Gonads	2.5×10^{-8}	1.0×10^{-8}	1.7×10^{-3}
Breast	1.6×10^{-8}	6.6×10^{-9}	1.1×10^{-3}
Red Bone Marrow	6.1×10^{-7}	2.6×10^{-7}	4.3×10^{-2}
Lung	2.8×10^{-8}	1.1×10^{-8}	1.9×10^{-3}
Thyroid	2.5×10^{-8}	1.0×10^{-8}	1.7×10^{-3}
Bone Surface	9.5×10^{-6}	4.0×10^{-6}	6.9×10^{-1}
Stomach	1.1×10^{-8}	4.4×10^{-9}	7.6×10^{-4}
Small Intestine	2.4×10^{-8}	1.0×10^{-8}	1.7×10^{-3}
Upper Large Intestine	1.5×10^{-7}	6.2×10^{-8}	1.1×10^{-2}
Lower Large Intestine	4.5×10^{-7}	1.8×10^{-7}	3.1×10^{-2}
Kidney	4.5×10^{-6}	1.9×10^{-6}	3.2×10^{-1}
CEDE	6.8×10^{-7}	2.8×10^{-7}	4.9×10^{-2}

[a] Sieverts (Sv) = 0.01 x rem

8.5.5 The CEC Environmental Radiological Monitoring Program

This section presents an overview of the proposed CEC radiological environmental monitoring programs for its preoperational and operational phases. A detailed discussion of these programs is presented in the NRC DEIS on the CEC (NRC, 1993a).

8.5.5.1 Preoperational Radiological Environmental Monitoring Program

The preoperational program will focus on collecting data needed to perform critical pathway analyses, including selection of nuclide/media combinations to be encompassed into the operational surveillance program. Identification of radionuclides will be performed using accurate and sensitive analytical equipment, as is technically appropriate. Data collection during this period will provide baseline information for evaluating any future changes in environmental conditions that might be caused by facility operation. The proposed preoperational program is somewhat more intensive than the operational program in order to provide this base of knowledge and to reflect changing conditions around the site as the facility is built, operated, and eventually decommissioned. This base of knowledge will provide adequate data to give assurance of the proper operation of containment and effluent controls, to support assessment of radiological impacts on the site environs, including potential impacts on members of the public, and to help determine compliance with applicable radiation protection standards.

This program will be initiated at least 2 years prior to the operation of the facility to provide a sufficient data base for comparison with, and provide experience to improve, the proposed operational radiological monitoring program (LES, 1993b). The NRC DEIS contains a discussion of the details of the program (NRC, 1993a).

8.5.5.2 Operational Radiological Environmental Monitoring Program

LES is committed to establishing a state-of-the-art environmental radiological monitoring program which supports its ALARA goal to minimize annual average concentrations of radioactive gaseous or liquid effluents at the boundary of the unrestricted area and beyond. The Operational Radiological Environmental Monitoring Program is basically a continuation of the preoperational program, and is discussed in detail in the NRC DEIS on the CEC (NRC, 1993a). As shown in the DEIS, the environmental media to be monitored, monitoring locations, and analytical LLDs are the same for both programs. The NRC staff concludes that the applicant's proposed LLDs for air, water, soil, and vegetation samples provide reasonable assurance of detection of background levels and are acceptable. With the exception of monitoring for airborne particulates and surface water, the frequency of sampling of other media will decline from quarterly (preoperational program) to semi-annually (operational program). The NRC staff reviewed the applicant's proposed action levels for alpha activity in air, water, and soil and concludes that these levels provide reasonable assurance that the CEC will maintain offsite concentrations below regulatory limits. The NRC staff finds that the applicant's proposed action level for alpha activity in vegetation may be high and thus recommends the following license condition:

> Notwithstanding the action level for gross alpha activity in vegetation in Table 5.2-2 of the applicant's Proposed License Conditions, the action level for gross alpha in vegetation collected in the environmental monitoring program shall not exceed 1.85×10^{-4} Bq/g (0.005 pCi/g).

8.6 Ensuring that Radiation Exposures are As Low As is Reasonably Achievable

NRC regulations require that occupational exposures and releases of radioactivity to the environs be as low as is reasonably achievable (ALARA).

There is currently no mandatory regulatory guidance on what constitutes ALARA for enrichment facilities. However, the general principles expressed in Regulatory Guides 8.10 (NRC, 1975) and 8.13 (NRC, 1987b) are applicable to occupational exposure control at the CEC Facility. The ANPR also requires that concentrations of radioactive materials from effluent releases at or beyond the exclusion area boundary during normal or accident conditions not create any undue risk to the health and safety of the public (NRC, 1988). This is reiterated in a draft Regulatory Guide (DG-8013, " 'ALARA' Levels for Effluents from Materials Facilities"; NRC, 1992c). The draft guide incorporates the elements of a Memorandum of Understanding (MOU) between the NRC and the EPA regarding non-reactor facility effluents that would constrain offsite doses to a member of the public to an ALARA goal of 10 mrem (TEDE) per year (NRC/EPA, 1992). EPA's 40 CFR Part 61 also sets 10 mrem per year as a requirement established by EPA.

CEC has committed to maintain doses to workers and members of the public ALARA, consistent with the new 10 CFR Part 20 ALARA requirements. The various aspects of facility operations designed to maintain occupational exposures ALARA are discussed in Sections 8.3 through 8.5. As discussed in Section 8.4, the operating experience of the three Urenco gaseous centrifuge plants in Europe, shown in Table 8.3, indicates that occupational doses at the proposed facility will meet the NRC ALARA requirements of 10 CFR Part 20, and the guidance of the ANPR.

In the case of assuring that offsite doses are ALARA, the NRC staff finds that the effluent monitoring program discussed in Section 7.5, and the environmental monitoring program discussed in Section 8.5.5, will provide assurance that releases and exposure of the public to those releases are maintained at an acceptable ALARA level.

9 NUCLEAR CRITICALITY SAFETY

A criticality accident is defined as the release of energy as a result of accidentally producing a wayward, or divergent, neutron chain reaction. Nuclear criticality safety is the protection against the consequences of an inadvertent nuclear chain reaction, preferably by the prevention of the chain reaction (ANSI/ANS-8.1-1983, [ANS, 1983]).

Natural uranium hexafluoride (UF_6), the feed material, and other chemical forms of natural uranium at the Claiborne Enrichment Center (CEC) are not capable of producing a wayward neutron chain reaction within the CEC facility. However, as the uranium is enriched in the U-235 isotope, the UF_6 and other chemical forms of uranium are capable of producing wayward neutron chain reactions. As the enrichment increases, the quantity of enriched uranium required to produce a neutron chain reaction decreases. Consequently, by establishing a safety basis for the maximum enrichment in the plant, a safety basis exists for all enrichments allowed in the plant.

Nuclear criticality safety begins with establishing criticality safety factors which are used to identify safe limits for process operations by reducing either critical mass quantities or critical dimensions of equipment. These safety factors must be incorporated into the original plant design and any changes made over plant lifetime. This section describes and evaluates the applicant's proposed administrative practices to identify and establish safety factors, and to incorporate them in the plant design. The administrative practices and the safety factors are incorporated into the license. The plant design is reviewed to verify the adequacy of the administrative practices and the safety factors.

This section of the Safety Evaluation report (SER) is based primarily on a review of the CEC Safety Analysis Report (SAR) (LES, 1993a), Revision 18, especially Section 4.5, "Nuclear Criticality Safety"; Section 6.3, "Enrichment and Other Processing Systems"; and Chapter 11, "Management Organization, Testing, and Operating Programs"; the "Criticality Safety Engineering Report, Revision 6" (CSER) (LES, 1993i); and the Louisiana Energy Services (LES) "Proposed License Conditions" (PLC), Revision 6" (LES, 1994).

9.1 Nuclear Criticality Safety Administration

The NRC staff's position is that an effective nuclear criticality safety program relies on a technically sound engineering design. Of equal significance for safe operation are administrative practices which ensure not only a safe design, but also maintenance and improvements of the design safety basis during operations over plant lifetime. This section examines the proposed organization and administrative practices to ensure safe operation during plant lifetime.

9.1.1 Plant Organization

The applicant states in SAR Chapter 11 that the CEC Manager, who reports to the LES President, has responsibility for operating the facility in a safe manner. The CEC Manager is responsible for the protection of the facility staff and the public from radiation and accidents, and is responsible for compliance with the license. The Quality Assurance (QA) Manager reports to the CEC Manager and is responsible for implementing the QA program. Five superintendents report to the CEC Manager and may act for the manager in his absence. The Operations Superintendent reports to the CEC Manager and is responsible for directing day-to-day operations. The Integrated Scheduling (IS) Superintendent reports to the CEC Manager and is responsible for directing the scheduling of enrichment operations. The Maintenance Superintendent reports to the CEC Manager and is responsible for directing and scheduling maintenance activities to maintain the facility in proper operating condition. The Compliance Superintendent reports to the CEC Manager and is responsible for directing activities to ensure that the facility remains in compliance or conformance with applicable regulations or codes. The Technical Support (TS) Superintendent reports to the CEC Manager and is responsible for providing support in the areas of health physics, chemistry, industrial safety, and engineering, which includes criticality safety. Reporting to the TS Superintendent are the Health Physics Manager, the Projects Manager, the Chemistry Manager, the Industrial Safety Manager, and the Performance Manager.

9.1.2 Nuclear Criticality Safety Organization

The Projects Manager is responsible for implementing facility modifications and reviewing facility procedures and modifications for nuclear criticality safety. The Projects Manager's group includes an individual who performs analyses or who reviews and approves the analyses, and conducts quarterly nuclear criticality safety inspections. When analyses are performed, a second individual will be added to perform the analyses or the independent review.

9.1.3 Position Qualifications

The applicant discusses the minimum education and experience requirements for the above positions (SAR Chapter 11). In the PLC, the applicant commits to requirements which are summarized as follows:

The CEC Manager holds a BS degree or equivalent in an engineering or scientific field and has 6 years' responsible nuclear experience. All superintendents have a BS degree or equivalent in an engineering or scientific field and 4 years' appropriate nuclear experience.

The Projects Manager has a BS degree or equivalent in engineering or science and a minimum of 3 years' appropriate nuclear experience. In addition, the Projects Manager has at least 1 year of direct experience in the administration of criticality safety reviews.

One projects individual has a BS degree or equivalent in engineering or science and at least 1 year's experience in implementing a criticality safety program. If facility or process changes require new nuclear criticality safety analyses, an individual with a BS degree or equivalent, who is trained in the physics of criticality, performs the analyses. A second trained individual, with a BS degree or equivalent and at least 2 year's experience in performing analyses and implementing nuclear criticality safety programs, independently reviews and approves the analyses.

As used in the applicant's documents, equivalent means higher education in another country which is equivalent to USA requirements for a BS degree. The resumes of the persons filling the above identified positions will be reviewed before receipt of licensed material as part of the required preoperational inspection process.

9.1.4 Safety Committee

The Facility Safety Review Committee (FSRC) functions include technical and administrative reviews and audits of authorized facility activities which may affect plant workers and public safety. The FSRC's responsibility includes reviews of ongoing and proposed nuclear criticality safety activities and practices. Nuclear criticality safety investigation, audit, and inspection reports will be included in the reviews. In addition, the FSRC will conduct an annual audit of the nuclear criticality safety area.

The FSRC will meet quarterly during initial operations. When stable operations are reached, the committee will meet at least three times a year, with a maximum interval of 6 months between meetings. The committee reports to the CEC Manager, who appoints at least five CEC technical or LES corporate staff members. Members have an engineering or scientific degree and 3 years' nuclear experience. The FSRC includes a person having, as a minimum, the qualifications of a nuclear criticality analyst. Proceedings, findings, and recommendations are provided in writing to the CEC Manager and to the responsible superintendent. Records of FSRC activities are maintained for the life of the facility.

9.1.5 Nuclear Criticality Safety Training Programs

The applicant describes an extensive training program in SAR Section 11.3. The program includes requirements for nuclear criticality safety training and will be developed to provide formal training in order to establish knowledge foundations and on-the-job (OJT) training to develop work performance skills. The nuclear criticality safety training program will meet the requirements of ANSI/ANS-8.20-1991 (ANS, 1991).

Newly hired people receive nuclear safety training in criticality safety before receiving unescorted access to the facility Controlled Access Areas (CAA). The Projects Manager certifies the training instructor. People with CAA access are given annual refresher training. OJT is designed to provide employees with the job-related skills and knowledge to perform a specific task. The qualification program includes the task and related procedures.

9.1.6 Plant Procedures

The applicant describes the plant procedures for pre-operational and operational testing in SAR Section 11.2. The applicant commits to using written procedures for all safety-related operations in SAR Section 11.4 and to having procedures to ensure that all criticality safety activities are carried out in accordance with written procedures. The procedures include limits on parameters to be controlled and corrective measures to return a parameter to its normal control band. The revised PLC contain these commitments.

In addition to operating procedures, the NRC staff finds that activities of the nuclear criticality safety function need to be performed in accordance with written procedures. The revised PLC, dated December 3, 1993, contain such a commitment.

9.1.7 Nuclear Criticality Safety Audits and Inspections

Annual audits of criticality requirements are conducted by a senior member of the projects group, according to SAR Section 11.4.4. The audits are performed in accordance with procedures approved by the CEC Manager. Records of the audit findings and corrective actions are maintained for at least 2 years. In the PLC, the applicant commits to annual audits and semiannual inspections. The NRC staff finds this commitment satisfactory.

9.1.8 Nuclear Safety Analyses for Facility or Process Changes

The applicant expects to make some equipment changes during the life of the plant. The applicant describes the process of making these changes in SAR Section 11.4.6. It is also necessary to ensure that the original CEC process description and safety analysis are maintained in a configuration control program. In the revised PLC, the applicant has committed to a configuration control program which requires that, for all possession, use, and storage activities with enriched uranium at the facility, LES will maintain written records of: (1) the current description of all enriched uranium processes at the facility; (2) a current identification of potential criticality accidents identified by a systematic hazards analysis process for all current activities; (3) for each of the potential criticality accidents identified above, a current safety analysis identifying all necessary limits on parametric controls to prevent an inadvertent critical configuration; and (4) administrative requirements to ensure that the engineered systems to limit the parametric controls are installed, maintained, and operated as designed. For each potential criticality accident identified in (2), LES will implement and maintain independent engineered or administrative controls so that the double-contingency principle of ANSI/ANS-8.1 is satisfied. In addition to identifying the limits and controls in (3), LES will document the requirements for maintenance, surveillance, personnel training, posting, and control of written procedures to ensure the effectiveness of the limits and controls.

9.1.9 Event Investigations

The applicant describes an internal program for reporting unusual events to the Licensing Manager and the Compliance Superintendent. Unusual events potentially threaten or weaken the effectiveness of the nuclear criticality safety program. The Superintendent determines the level of investigation and any external reporting requirements. Lessons learned are documented as part of the corrective actions.

9.1.10 Staff Evaluation of Nuclear Criticality Safety Administration

The NRC staff has evaluated the proposed organization and administrative programs for establishing, implementing, and maintaining the nuclear criticality safety program. The PLC provide reasonable assurance that the licensed special nuclear material (SNM) can be possessed and used without undue risk to the public.

9.2 Nuclear Criticality Safety Criteria and Safety Margins

The applicant discusses the design basis for nuclear criticality safety in CSER Section 2.1. This section reports the applicant's considerations and NRC staff's comments on each of them. These comments are the basis for further evaluation of safety criteria and margins in this report.

Consideration 1. The feed material can go critical only under special and carefully controlled conditions which are not possible at CEC. The NRC staff agrees with the applicant. Natural uranium fuel elements can become critical under special circumstances such as in heavy water (D_2O) moderated reactors. These special circumstances do not exist at the CEC.

Consideration 2. The depleted uranium can never go critical under any circumstances. The NRC staff agrees that this consideration is true for depleted uranium tails between 0.2 and 0.34 weight percent (wt %) U-235. For slightly depleted material, however, the NRC staff position on Consideration 1 is applicable.

Consideration 3. The feed and depleted uranium constitute the bulk of the material handled. The NRC staff agrees with this statement; however, significant quantities of enriched uranium are handled. A detailed safety evaluation is provided below.

Consideration 4. The enrichment process is carried out under vacuum. The NRC staff agrees that, as discussed below, the vacuum process contributes significantly to the safety of the enrichment process by preventing the release of uranium and the intrusion of moderating materials.

Consideration 5. The quantity of uranium in the process equipment is small. The NRC staff agrees with this consideration as it pertains to the centrifuges and cascade halls. However, significant quantities are handled in the product loadout areas and, perhaps, the Technical Services Area (TSA).

Consideration 6. Impurities (potential moderators) must be avoided because of other process considerations. The use of moderators such as water and hydrocarbon oils in the UF_6 process must be avoided. The NRC staff agrees that, in certain process areas, the use of moderators such as water and oil must be controlled. In all UF_6 areas, hydrocarbon oils must be controlled for chemical safety.

Consideration 7. Wastes produced by maintenance, off-gas treatment, and other means contain only small quantities of uranium. Although the NRC staff agrees that this result is the normal situation, because such wastes are collected in nonfavorable geometries, a safety evaluation is required. Moreover, the safety of potentially large quantities of waste from off-normal conditions is evaluated.

Consideration 8. The product is of low enrichment and is collected in commonly used, internationally accepted cylinders specifically designed for that purpose. The NRC staff agrees that the specified cylinders are acceptable for processing, storage, and shipping moderation-controlled UF_6 of not more than 5 wt % U-235. The NRC staff's evaluation of controls on the cylinders to preclude moderators and higher U-235 enrichments is presented below.

Consideration 9. Cold air (rather than water) is used for desubliming the product UF_6 into the 30B cylinders. The NRC staff agrees that this design significantly reduces the risk of an interaction between UF_6 and a water moderator in the product handling area and thereby reduces the risk of criticality.

Consideration 10. The enrichment level is set by controls under administrative procedures and cannot easily or accidentally be changed or go undetected. The NRC staff's evaluation of the controls and administrative procedures which limit or detect enrichment levels above 5.02 wt % U-235 is presented below.

9.2.1 Administrative Practices

The applicant commits in the PLC to a modification of the double-contingency principle as stated in ANSI/ANS-8.1. The modified principle requires that "process design shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes before an accident is possible." This modification is a significant improvement and is acceptable to the NRC staff.

The applicant's general approach to nuclear criticality safety is to prevent enrichment excesses, use favorable geometry equipment when practicable, provide moderation control within the UF_6 enrichment process, and use strict mass control on solutions. This hierarchy of controls is acceptable to the NRC staff.

The applicant commits to certain documentation of safety analyses and design practices, that is, document analyses and the control of design changes in accordance with written procedures. The applicant commits to using written operating procedures, posting nuclear criticality safety limits, and performing pre-operational testing and inspection. These commitments, combined with the commitments in Section 9.1.8 above, describe a program acceptable to the NRC staff.

9.2.2 Single Unit Safety Factors

In SAR Section 4.5, the CSER, and the PLC, the applicant provides nuclear criticality safety factors for process equipment, that is, single units. These factors, taken from Regulatory Guide (RG) 3.52, Rev. 1 (NRC, 1986b), are applied to critical geometric dimensions or masses. The mass/geometric units and the proposed CEC safety factors are:

Mass/Geometry	Safety Factor
Volume	1.34
Cylinder Diameter	1.12
Slab Thickness	1.18
Mass*	1.34
Mass	2.23

*Only when double batching is not possible because of physical volume restrictions.

The NRC staff has confirmed that these safety factors are consistent with the recommended safety margins of RG 3.52, Rev. 1.

The safety factors are applied to critical dimensions or mass. The critical dimensions reported in the SAR were determined by the NRC staff to be critical or slightly subcritical. The NRC staff calculated the k-effective (k_{eff}) values for the critical units specified in PLC, Table 4.2-1, and confirmed that the specified units are near critical, but slightly subcritical, and that the safety factors in RG 3.52 ensure subcriticality for the individual units. The NRC staff also calculated the k_{eff} for the safe units with full-water reflection by using the XSDRNPM code and a 27-group cross-section in a personal computer (PC) version of the SCALE code for the calculations. Several calculations were made to validate results with experimental benchmark data and with the controlled SCALE code at ORNL. The results for UO_2F_2 solutions with uranium enriched to 5.02 wt % U-235 are shown in Table 9.1:

Table 9.1 K_{eff} for critical and safe parameters

Parameter	Critical Value	k_{eff}
Volume	25.0 l	0.95
Cylinder Diameter	24.6 cm	0.95
Slab Thickness	12.4 cm	0.98
Mass	35.5 kg U	0.99
	Safe Value	
Volume	18.6 l	0.90
Cylinder Diameter	21.9 cm	0.90
Slab Thickness	10.5 cm	0.92
Mass	26.4 kg U*	0.94
Mass	15.9 kg U	0.86

* Double batch not possible

The reduction in critical size is only one of many safety considerations. Another consideration is stability of the geometric units. If the geometric unit increases in size because of corrosion, pressure, temperature, and the like, the favorable geometry unit could have a reduced margin of safety or become unsafe, that is, k_{eff} could be equal to or greater than 1.0. The applicant commits to design requirements to ensure that the integrity of geometric units is maintained by design and testing.

9.2.3 Arrays of Safe Units

In addition to controlling the size of each individual unit in the facility, the neutron interaction between units must be controlled so that the array of safe units in the facility does not become critical. This control is achieved by ensuring sufficient spacing between favorable geometry or safety mass units so that neutron leak out of the fissile material is captured by non-fissionable materials (neutron poisons). The applicant does not discuss safety criteria for arrays of favorable geometry or safe mass units. However, in SAR Table 4.5.1, the applicant provides a reference to an empirical solid-angle method for evaluating unit neutron interaction. The referenced solid-angle method is acceptable to the NRC staff.

The applicant also used MONK, a Monte Carlo code comparable to KENO, which has been developed by UKAEA, to calculate k_{eff} for an array of product cylinders. However, the applicant provided no validation for this calculational method. Accordingly, it is necessary to limit the use of the MONK code until validation is established. In the revised PLC, the applicant agrees that, before using the MONK code for CEC process design changes made without NRC approval, the licensee will provide a summary report on the validation of the MONK code. The report will specify the range of applicability and other parameters specified in ANSI/ANS-8.1. The validation report will be maintained at the CEC facility.

9.2.4 Special Considerations

The applicant proposes to use limited and controlled moderation as another nuclear criticality safety control. This moderation control is the basis for safety of UF_6 in the product loadout Model 30B cylinders. SAR Table 4.5-2 shows that the applicant limits the amount of hydrogen so that H/U (hydrogen-to-uranium atom ratio) will not exceed 1 or that the amount of hydrogen is less than 2.0 kilograms. This dual limit is necessary to control conditions which can exist in a partially filled cylinder. The NRC staff agrees that this limit is acceptable.

Independent staff calculations using a KENO V.a code in a PC version of SCALE verify that k_{inf} (k_{eff} for an infinite mass of UF_6 or an infinite array of UF_6 cylinders) for moderation-controlled uranium enriched to 5.02 wt % U-235 can exceed unity. The results of these NRC staff calculations are shown in Table 9.2:

Table 9.2 K_{inf} for arrays of moderated UF_6

Material	H/U Ratio	Interspersed Water gm/cm^3	K-inf±sigma
UF_6	0	0	0.7086±.0016
	0.3	0	0.8441±.0024
	1.0	0	1.0452±.0030
UF_6 in infinite cubic array of cylinders	0	0	0.6891±.0017
	0.3	0	0.8144±.0022
	1.0	0	1.0038±.0031
UF_6 in infinite planar array of cylinders	0	0	0.4794±.0018
	0.3	0	0.6113±.0028
	1.0	0	0.8355±.0038
	0	0.05	0.5638±.0026
	0.3	0.05	0.6642±.0029
	1.0	0.05	0.8544±.0034
	0	1.0	0.5218±.0024
	0.3	1.0	0.6296±.0031
	1.0	1.0	0.8336±.0039

The NRC staff agrees with the applicant that limited and controlled moderation for an infinite planar array of cylinders, with or without interspersed moderator, is an acceptable and necessary nuclear safety control. This table shows that cuboid arrays are not safely subcritical. However, the applicant specifies that storage is limited to one-high planar arrays.

Nuclear criticality safety for other processes relies on administrative control. Procedures control the maximum enrichment of 5.02 wt % U-235. In SAR Section 4.5.1, the applicant claims that excessive enrichment is prevented by plant and equipment features. However, in SAR Table 4.5-2, the applicant recognizes that administrative controls are needed to adjust and control plant equipment to obtain the desired enrichment and to prevent excessive enrichment.

9.2.5 Staff's Evaluation of Nuclear Criticality Safety Criteria and Safety Margins

The NRC staff concludes that the applicant's proposed nuclear criticality safety criteria and safety limits are adequate to provide reasonable assurance that the health and safety of the public and of nuclear workers are adequately protected.

9.3 Staff Analysis of Applicant's Proposed Nuclear Criticality Safety Criteria for Plant Design and Operation

This section briefly describes the steps in the applicant's proposed process, summarizes the applicant's safety bases, and provides the NRC staff's evaluation of the applicant's nuclear criticality safety program. The evaluation is based on LES criteria and commitments in the SAR, the CSER, the License Application, and the suggested contents in RG 3.52, "Standard Format and Content for the Health and Safety Sections of License Applications for Uranium Processing and Fuel Fabrication."

9.3.1 Natural UF_6 Receiving and Storage

The UF_6 handling activities consist primarily of receipt, storage, transport, and vaporization of UF_6 in shipping containers. The vaporization step includes offgasing the cylinder before introducing feed into an enrichment cascade.

In CSER Section 2.1, the applicant notes that natural UF_6 cannot become critical during any handling or processing operations authorized at CEC. As discussed above, the NRC staff agrees that no nuclear criticality safety analysis is needed for handling natural UF_6 at CEC.

9.3.2 Enrichment Centrifuges and Cascades

In this section of the SAR/CSER, the applicant deals with the individual centrifuges and the array of centrifuges inside the cascade halls. Support systems, such as the contingency dump system and the product loadout system, are discussed below. The uranium in all centrifuges is normally in the form of gaseous UF_6 under negative pressure. As a result, only a few grams of uranium are in a centrifuge at any one time. Each centrifuge is monitored by a

Cascade Protection System, which either initiates corrective action or allows operators to take corrective action.

The abnormal operations which the NRC staff or the applicant has evaluated include:

(a) Enriched feed material

(b) Enrichment of the feed material above 5.02 wt % U-235

(c) Loss of containment of the UF_6 gas

(d) In-leakage of water or other moderator

(e) Accumulation of enriched uranium in unfavorable geometry equipment.

In abnormal operation (a), feed material is received in either 10-ton or 14-ton cylinders. Product cylinders are limited to the 2.5-ton cylinder. Because of the significant difference in size and the differences in engineered systems, the possibility of interchanging product and feed cylinders is negligible. However, the feed cylinders can be used for containment of enriched uranium. Accordingly, the applicant must verify that the incoming feed material is not enriched. The applicant addresses this issue in the proprietary Fundamental Nuclear Material Control Plan. As proposed, the plan provides assurance that enriched uranium will not inadvertently be introduced into the process.

Abnormal operation (b) is the enrichment of normal feed material above 5.02 wt % U-235. During startup operations, the enrichment performance of each cascade hall is benchmarked against the controlling computer program. Then, whenever an enrichment campaign is established, the computer program monitors the controllers which control the enrichment. A second individual verifies the manual manipulation of the controllers. Gas samples of the UF_6 product are taken within 24 hours to confirm the new enrichment level. If necessary, the controllers are readjusted to yield the desired enrichment. Samples are again taken to confirm the proper enrichment. The Cascade Protection System monitors pressure, temperature, and valve positions for cooling water, feed header, product take-off, and tails take-off systems and power for the centrifuges. Once an enrichment campaign is established, the monitoring data is used to verify that the enrichment level does not change.

The NRC staff agrees that the Cascade Protection System provides an adequate control system for enrichment control. While slightly higher enrichments might occur during an enrichment change, the quantity would be small and of no nuclear criticality safety consequence. The routine monitoring of the product provides further assurance that enrichments above 5.02 wt % U-235 do not occur.

Abnormal operation (c), loss of containment of the UF_6 gas, does not provide a significant criticality hazard or risk because the density of the gas under negative pressure is very low, and only a few grams of uranium are in any centrifuge. Thus, the release of UF_6 would have to occur from many centrifuges before a critical mass of uranium would be released. If containment were lost while the containment was under negative pressure, in-leakage of moist air would occur. The moisture would react with the UF_6, and the resulting UO_2F_2 would settle out in small quantities in each centrifuge. The increased pressure in the cascades would cause control room alarms but would not have any criticality safety significance. The NRC staff agrees that nuclear criticality resulting from loss of containment of UF_6 gas is extremely remote.

Abnormal operation (d), the in-leakage of water or other moderator, results in the shutdown and isolation of the centrifuge or centrifuges because of the sudden increase in pressure from the UF_6-H_2O reaction. Because of the small quantity of UF_6 in any one centrifuge and the favorable geometry of piping in the cascade halls, the potential in-leakage of water is of no significance to nuclear criticality safety.

Abnormal operation (e), the accumulation of enriched uranium in nonfavorable geometry equipment cannot occur because there are no nonfavorable geometry pieces of equipment in the cascade halls. The centrifuges and all process piping of UF_6 into, through, and out of the cascade halls are favorable geometry. The number and size of pipes in pipe runs are controlled so that a large nonfavorable geometry vessel consisting of many small pipes is not installed during plant construction. Control of multiple pipes is discussed later. Accordingly, abnormal operation (e) is not a safety concern.

In the revised PLC, the applicant has made the following commitments for the centrifuges, cascades, and assay units:

- Limit the maximum inside diameter of the centrifuge housings to the safe diameter specified in PLC, Table 4.2-1

- Introduce no enriched uranium as feed material to the cascades

- Provide control room monitoring of cascade header pressure, power, feed supply pressure and valve position, cooling water temperature and flow, product take-off system pressure and valve position, and tails take-off system pressure and valve position. For unusual monitoring results, procedures shall describe corrective action including shutdown requirements

- Provide no interconnections allowing enriched uranium to flow to the feed system

- Limit the maximum uranium enrichment in any cascade to 5.02 wt %

- For each authorized enrichment campaign, calculate the control valve settings and instruct the operators, in writing, of the correct valve settings. The new valve settings will be independently confirmed and recorded. If resulting pressures do not compare with predicted pressures, initiate corrective actions

- For each new campaign, analyze the UF_6 product for enrichment within 24 hours, and compare the results with predicted results. If an adjustment is required, repeat the actions in the preceding step.

The NRC staff agrees that these limits and controls are adequate.

9.3.3 Product Take-Off System

The applicant states that incoming empty product cylinders are transported on a rail-mounted transporter inside the facility. Each cylinder is weighed, inspected for contaminants, and evacuated. As needed, empty product cylinders are transported to product take-off stations and connected to process piping. Any process piping exposed to air is evacuated by a mobile vacuum pump.

During normal operations, enriched gaseous UF_6 is continuously withdrawn from the centrifuge cascades. Vacuum pumps are used to move the UF_6 to product cylinders where the UF_6 is solidified. The product cylinders are located on a scale in an air-cooled cold chest. The fill weight is controlled to prevent cylinder rupture during product sampling.

Each filled product cylinder is placed inside an autoclave, connected by manifold to a product sample bottle, and heated. After the sample is drawn, both the autoclave and sample bottle are cooled by chilled water so that the UF_6 is solidified before movement outside the autoclave.

An optional process step is use of the Product Blending System, in which the contents of two product containers can be combined to provide a specified U-235 assay. Each donor cylinder is heated in an autoclave so that the contents can be transferred to a "receiver" product container in an air-cooled cold chest. The empty "donor" cylinder and the filled cylinders are then moved to the product storage area.

Initially, when the product cylinder is being filled, gases in the product or process lines are vented through the product vent system. The autoclaves for product sampling are vented to the Gaseous Effluent Vent System (GEVS). The HF monitoring system is designed to detect UF_6 leaks. Light gases in cylinders in the Product Blending Station are purged to the blending vent system.

The applicant identifies three abnormal operations which could affect nuclear criticality safety. These three operations are the presence of moderating material in the empty product cylinders, the intrusion of moderating material into the cylinder while it is being filled, and the production of enrichments above 5 wt % U-235.

Product cylinders containing UF_6 enriched to 5.020 wt % U-235 are criticality-safe if moderators are limited and controlled. The applicant proposes in-process limits on hydrogen moderators so that the H/U atomic ratio does not exceed 1.0 or, if the H/U ratio is greater than 1.0, the cylinder contains less than 2 kilograms of hydrogen. In addition, the applicant must comply with transport requirements which, depending on the form of the moderator, may require a lower H/U ratio.

The product cylinders do not satisfy the double-contingency principle. If a filled cylinder at ambient temperature were punctured, enough water could enter the cylinder and moderate enough enriched uranium to form a critical mass. Avoiding this occurrence requires that special handling equipment be used by trained operators. The applicant is committed to using such handling equipment as described in the SAR and the PLC.

The applicant's bases for these limits are calculations performed by British Nuclear Fuels, Ltd., to justify not only the individual product cylinder but also a planar array of similar cylinders in contact. The applicant claims that such an array is subcritical under all weather conditions. In the CSER, Appendix E, the applicant provides information on the calculations. The applicant also references calculations from the literature, but such calculations are not readily applicable because of differences in array size, enrichment, or degree of moderation.

The NRC staff also used the SCALE program to perform a series of calculations with the results shown in Section 9.2.4. Based on these calculations, the NRC staff agrees that the H/U atomic ratio of not more than 1.0 provides an adequate margin for subcriticality of product cylinders in one-high planar arrays.

The NRC staff has examined the applicant's procedural steps taken to ensure that excessive moderator material is not present in empty product cylinders. The steps include two separate weighings, a vacuum test, a visual examination, and a high-pressure indication upon initiation of filling.

The second abnormal operation condition is the intrusion of moderating material into the cylinder during filling. This scenario appears unlikely because the only process water in the cylinder filling process is chilled water in coils which are outside of the autoclave and are structurally independent of it. In addition, introduction of moisture would result in a high-pressure situation because of the reaction with UF_6. This high pressure would be detected immediately by the process monitoring system and lead to process shutdown.
The third abnormal condition is the production of enrichments above 5 wt % in the U-235 isotope. Although this condition is dealt with extensively in the Material Control and Accounting (MC&A) section, it is reviewed here. The NRC staff considered two conditions which could lead to unauthorized high enrichments, that is, the initial improper instrument

and control settings, and the long-term drift of instrument and control settings. The applicant plans to monitor the enrichment for each assay unit daily during an enrichment change. Adjustments are made as necessary to the instruments and controls, and to the computer model which is used to predict and control the enrichment process. In addition, each product cylinder is sampled as it is removed from the fill station to provide an additional check on enrichment.

In the revised PLC, the applicant has committed to the following actions for the product take-off system:

- Verify by cylinder weight, baroscope inspection, and vacuum testing that no internal contaminants are present before an empty cylinder is brought to a product take-off station

- Limit the contents of each product cylinder to the authorized fill limit using of the continuously monitored load cell system. Before initial use of the load cell system, develop and implement a maintenance and test program

- Vent volatile gases to the Product Vent System. Conduct investigations and take corrective actions whenever a predetermined number of vent cycles per cylinder have occurred

- Solidify the UF_6 only with cool air during and after filling cylinders in the product filling station and before moving the cylinder

- Heat product cylinders for blending or sampling only in autoclaves and only with electrically heated air. The autoclave pressure and temperature shall be automatically controlled, continuously monitored, and alarmed during the heating cycle. After blending or sampling, but before moving the cylinders, solidify the UF_6 by circulating water in the structurally independent cooling coils external to the autoclave

- Limit the H/U ratio to not more than 1.0 or limit the hydrogen content to two kilograms

- Store filled product cylinders only in one-high planar arrays

- Transport filled product cylinders only with CEC-approved rigging and transport vehicles.

The NRC staff agrees that the controls are adequate for ensuring a safe margin of subcriticality. The off-gas systems, which need to be evaluated, are considered in a subsequent section.

9.3.4 Desublimers

The applicant's process requires desublimer systems to remove gaseous impurities from UF_6 cylinders. Desublimers are used to vent feed, product, product blending, and tails cylinders. Each system consists of a cold-trap vessel, chemical traps, and a vacuum pump. Each cold trap has external closed-loop heating and cooling coils which use Freon as the heat-exchange medium. As discussed above, nuclear criticality safety is not an issue with feed or tails cylinders which are both vented to the Feed Purification Desublimer. The feed desublimers are isolated from product and product blending systems. Thus, the feed desublimers are not discussed further.

The light gases removed by the product desublimers include air, HF, reaction products of atmospheric moisture and UF_6, and other gases. These gases are removed from the product cylinder when the cylinder reaches a specified pressure, that is, about 45,000 pascals (6.5 psia). At this pressure, the product cylinder is isolated from the process and vented into the cold trap (desublimer) which, in turn, is isolated from the ventilation system.

When the desublimer reaches 5,000 pascals (0.7 psia), the desublimer is isolated. The UF_6, which typically amounts to 1 kilogram per venting cycle, is solidified in the desublimer. The remaining gases are vented through the chemical traps, which remove HF, trace quantities of UF_6, and vacuum pump lubricant before venting to the GEVS.

The single-tube product desublimer vessel is 40.6 centimeters (16 inches) in diameter and is capable of holding 3,700 kilograms of UF_6. The applicant proposes to operate each desublimer with an administrative limit of 100 kilograms of UF_6. This administrative limit is based on an assumed 4 kilograms of UF_6 carryover for each venting cycle. When the limit is reached, the desublimer is heated, and the UF_6 is transferred to a product cylinder.

The desublimer is a nonfavorable-geometry vessel because of its size, that is, 40.6 centimeters (16 inches) in diameter by 6 meters (17 feet) in length. The applicant has based nuclear criticality safety on moderation control which, if maintained, ensures an adequate margin of safety.

The applicant has calculated that the desublimer would contain less than a safe mass with not more than 2.4 kilograms of hydrogen in the desublimer. For the abnormal condition, the applicant does not have any means of limiting the intrusion of water to 2.4 kilograms or of determining the amount of hydrogen in the desublimer except for the measurement of pressure in the desublimer. The 2.4 kilograms of hydrogen would cause an immediate increase in pressure reading if the hydrogen were in the form of HF; if the hydrogen were in some other chemical form such as water, the hydrogen would react with UF_6, create gaseous products such as HF as well as solid UO_2F_2, and cause increased pressure in the desublimer. Thus, both the pressure and pressure changes need to be monitored closely. In the CSER, the applicant agrees to periodically calibrate and functionally test the monitoring and control system which monitors pressure and pressure changes in order to detect the presence of moderator. To control the buildup of UO_2F_2, on a quarterly basis (maximum interval of 4 months), LES will inspect the product and product blending desublimers to verify that only limited solid uranium compounds remain in the desublimers after the UF_6 is transferred to the product cylinders. If uranium solids are present in excess of 25 kilograms, the material shall be removed before reuse of the desublimer system.

In the revised PLC, dated December 3, 1993, the applicant has made the following commitments for each product and product blending desublimer system:

- Use only chemical traps, pumps, and piping which satisfy the geometric limits specified in PLC, Table 4.2-1

- Maintain a positive pressure nitrogen atmosphere between the inner desublimer vessel and the outer shell. The positive pressure shall be continuously monitored and alarmed to detect loss of atmosphere

- Heat and cool the desublimer vessel by refrigerant in external closed-loop heating and cooling coils

- Fit each inlet and outlet pipe with two valves in series, one of which is automatically operated by signal from continuous monitoring pressure and temperature sensors

- Record the number and frequency of cylinder vents to control the desublimer inventory and to detect abnormal venting intervals. Investigate any abnormal venting

- Develop and implement a preventative maintenance and test program for the nitrogen pressure monitor alarm system, the pressure and temperature sensors and valve system, and the vent frequency and interval monitoring system.

The NRC staff agrees that these controls are adequate for nuclear criticality safety. The chemical traps, pumps, and piping are reported to be safe by favorable geometry. These systems, however, are discussed further below.

9.3.5 Chemical Traps For Mobile Pump Sets and UF_6 Cylinder Venting Systems

Chemical traps are used in mobile vacuum pump sets, UF_6 cylinder venting systems, and the Contingency Dump System. The traps are intended to remove HF, trace quantities of UF_6, and vacuum-pump oil vapors from off-gas streams being vented to the GEVS for further treatment. The chemical traps for the Contingency Dump System are discussed in the next section.

A mobile vacuum pump set consists of a trap with a layer of activated carbon and a layer of aluminum oxide, a vacuum pump, and an activated-carbon trap in series. The size of each component of the mobile pump sets is less than a safe-diameter cylinder. Neutron interaction between the several components of the mobile vacuum pump set is considered in CSER Appendix J, and an infinite array of pump sets is considered in order to allow for interaction with other process equipment. A number of conservative assumptions are outlined in Appendix J. The maximum k_{eff} for the system is calculated to be less than 0.93.

The NRC staff agrees that the component trap designs are favorable geometry units. The activated-carbon trap collects the UF_6, and, because of operating conditions, the HF is collected on the alumina trap. Only trace quantities of UF_6 are expected to pass through the carbon filters and be adsorbed on the alumina traps. The applicant's interaction analysis assumes that certain equipment used in existing plants would be used in the CEC facility. The NRC staff will confirm this assumption before plant operation by inspection.

The applicant has made the following commitments in the PLC for chemical traps for mobile pump sets :

- Limit the trap inside diameter to 18.5 centimeters
- Limit the pump free volumes to not more than 7 liters
- Limit the k_{eff} of each pump set to not more than 0.93
- On an annual basis (maximum interval of 15 months), weigh and replace the activated carbon in the chemical traps.

The NRC staff agrees that these controls are adequate for nuclear criticality safety.

9.3.6 Contingency Dump Traps

Contingency Dump Traps perform the same function as the chemical traps discussed above. However, because these traps are used only for the contingency of an abnormal vacuum, that is, excessive pressure, the design allows for the rapid dump of an entire cascade.

The chemical trap is filled with sodium fluoride (NaF) rather than activated carbon. This compound allows for the rapid pressure release of an entire cascade. The UF_6 and HF react with the NaF while other gases pass through the trap into a common assay header, which provides a secondary surge volume. Vacuum to the common header is provided by two vacuum pumps. Valve interlocks ensure that only one cascade surge vessel can be vented to the assay unit header at one time.

For maximum enrichment operating conditions of 5 wt % U-235 product and 0.34 wt % U-235 tails, the applicant has determined that the average cascade enrichment inventory will not exceed 1.5 wt % U-235 for normal or abnormal conditions. On this basis, the applicant calculated that the safe diameter for the NaF trap is 60.3 cm. This dimension was determined by applying a safety factor of 1.12 to the critical dimension for 1.5 percent U-235 enriched uranium solution. No credit was taken for the poison effect of the NaF. In the event that a dump occurs, trap contents will be discharged to safe containers and stored pending disposal.

The NRC staff did not verify that the average uranium enrichment in the cascades does not exceed 1.5 wt %. The NRC staff agrees that the trap is subcritical for an optimally moderated solution at an enrichment of 1.5 wt % U-235. The trap would also be safe with 5 wt % U-235 without moderation. The NRC staff cannot identify a condition in which 5 wt % U-235 and optimum moderation co-exist. Thus, the combination of total cascade inventory, the average enrichment in a cascade, the reduced density of UF_6 because of the presence of the NaF, and replacement of traps after an emergency dump, ensures that nuclear subcriticality is maintained in the event of an emergency dump. The vacuum in each trap is monitored to guard against loss of vacuum to protect the centrifuges.

For contingency dump traps, the following commitments from the SAR have been made in the PLC. The licensee will:

- Limit the inside diameter of the dump traps to not more than 51.7 centimeters

- Independently verify and document that the contingency traps have been loaded with the correct amount and type of NaF powder and pellets before use

- Replace the trap media immediately after an emergency dump.

The nuclear criticality safety of the vacuum pumps, surge volumes, and process piping is evaluated in subsequent sections.

9.3.7 Vacuum Pumps

Vacuum pumps are used throughout the uranium enrichment process to maintain the necessary process conditions; discharge product and tails into collection cylinders, evacuate equipment and piping for maintenance operations; and after exposure to the atmosphere, obtain samples, remove light gases from feed and product cylinders, and maintain a vacuum in auxiliary systems, for example, the Contingency Dump System. Under normal conditions, only the process pumps contact more than trace quantities of UF_6. Fomblin oil is used as a lubricant in the pumps to avoid the chemical reaction which can occur with hydrocarbons. The oil, which may become contaminated with impurities, is treated and reused. The treatment process is evaluated in Section 9.3.9.

The applicant asserts that criticality cannot occur within the pumps because the free volume is less than 14 liters. The safe volume for solutions with uranium enriched to 5 wt % U-235 is 18.6 liters. Spacing is such that the solid-angle criterion is not exceeded. The NRC staff agrees that a free volume of 14 liters is safely subcritical.

9.3.8 Process Piping

Process piping provides confinement for the UF_6 material and excludes moderating materials such as moisture. When the UF_6 is under positive pressure, autoclaves provide secondary confinement. Process piping is heat traced, and valves are located in heated enclosures (hot boxes) if the operating pressures exceed 5,000 pascals (0.7 psia).

The applicant has determined that all process piping inside diameters do not exceed 21.9 centimeters. Parallel piping for product material is not permitted unless the interacting pipes can fit into an envelope which does not exceed 18 centimeters in inside diameter. A safety analysis for pipe bends or intersections is in CSER, Appendix H. As expected, MONK calculations indicate that pipe intersections or bends must be reduced. To effect the SAR commitments for these reduced pipe diameters for pipe bends and pipe joints, the applicant has revised the PLC to require the licensee to limit inside diameters of process piping to not more than 21.9 centimeters. When the process requires pipe bends or pipe intersections, the piping shall fit inside a maximum 18-centimeter, inside-diameter envelope.

9.3.9 Fomblin Oil

Fomblin oil, which is used as a lubricant in vacuum pumps, gradually thickens as impurities accumulate in it. The primary impurities are UO_2F_2 and UF_4, which are the reaction products of UF_6. As the oil thickens, pumps are replaced by other installed pumps. For pump maintenance, the oil is drained from the pump and treated so that the oil can be reused. The removal of uranium compounds starts with the addition of Na_2CO_3 to form a $Na_24UO_2(CO_3)_3$ precipitate. After the mixture is heated, the precipitates are removed in a centrifuge. The solids are dissolved in citric acid and transferred to a collection tank in the TSA. The oil is mixed with activated carbon to remove any hydrocarbon impurities. A diatomaceous earth filter removes the resulting sludge which is sent to solid waste disposal. If the oil does not meet recycle specifications for uranium (50 parts per million [ppm]) and hydrocarbons (3 ppm), it is processed again.

For criticality safety control, the applicant places the used Fomblin oil in containers which are adequately subcritical by volume control. The bottles of used oil are stored in a planar array. Processing of the oil is carried out by limiting the used oil to one safe mass of uranium (15.9 kilograms U). Wastes are sampled before transfer to unfavorable geometry containers. No criticality controls are placed on the recovered Fomblin oil. In the revised CSER and PLC, the applicant has committed to visually inspecting and sampling the recovered oil to ensure that the contaminants have been removed.

9.3.10 Contaminated Solid Wastes

The applicant's process results in the production of solid waste from different plant activities. Wastes are categorized as wet solids or dry solids. Dry radioactive solids contain less than 1 volume percent as liquids. All wastes are taken to the Radioactive Waste Storage Area where the wastes are inspected. Nonradioactive waste is segregated from radioactive waste thereafter.

Wet radioactive wastes are expected to be wet trash from decontamination operations, sludge from Fomblin oil recovery operations, and centrifuged precipitate from the citric acid treatment process. The applicant expects to transfer these wastes either to a licensed waste processor for solidification or volume reduction or to a licensed disposal facility.

Dry radioactive solids are expected to result from decontamination operations; spent activated carbon, aluminum oxide, and sodium fluoride; pre-filters and high efficiency particulate air (HEPA) filters; Liquid Waste Disposal System (LWDS) dryer operations; and miscellaneous materials. These wastes are sampled for uranium content and placed in storage or shipping containers for transfer to a licensed waste treatment or a disposal facility. Metallic contaminated wastes may be decontaminated for disposal as industrial waste.

The applicant's nuclear criticality safety program for radioactive solid waste consists of a program to limit each container to less than 4.6 kilograms of uranium and an area density of less than 4.6 kilograms of uranium per square foot. The applicant derives this value from LA-10860-MS, Figure 17 (Paxton, 1986). The critical area density is decreased by a safety factor of 2.3 to arrive at the proposed storage limit.

The NRC staff has evaluated these limits against the surface density equation in TID-7016, Revision 2 (Thomas, 1978), and finds them to be conservative. It is necessary to specify a minimum 1-foot edge-to-edge spacing between containers, however, in order to limit neutron interaction. In the revised PLC, the applicant commits to maintaining a minimum 30-centimeter, edge-to-edge spacing between units spaced by the surface-density method.

9.3.11 Contaminated Liquid Wastes

To serve different process and process areas, the applicant designed six primary liquid waste collection systems, each containing one or two contaminated waste liquid collection tanks. After collection and sampling of the liquids, the liquids and solid contaminants are separated by converting the liquids to steam in a dryer. The solids are removed daily for disposal as radwaste. The steam is condensed, sampled, and released as liquid effluent. The safety basis for each system is discussed separately.

<u>Effluent Collection Tanks - Units 1, 2 and 3</u>

There are two collection tanks for each of the three plant units. Each tank has a capacity of 4,012.5 liters (1,060 gallons) and is located in the effluent pit. Water from floor drains, area washdowns, and utility sinks is collected by gravity drain lines. No decontamination operation discharges liquids to this system.

Except for product sampling or blending, large quantities of UF_6 are in solid form under vacuum. Loss of containment would cause the slow sublimation of the solid UF_6; the resulting gaseous UF_6 would immediately react with atmospheric moisture to form solid UO_2F_2. These solids would have to be washed into the effluent tank drains. Special administrative procedures are used for such cleanup operations.

Product sampling and blending cylinders are in autoclaves whenever the cylinders are heated or contain liquid UF_6. Each autoclave provides secondary containment. The autoclave door is interlocked with instrumentation capable of detecting the pressure rise associated with a UF_6 release. The door is not opened after a UF_6 release. After abnormally high pressures are reduced to normal by the vacuum in the ventilation system, the door can be opened, and residual contamination can be removed.

The applicant's basis for nuclear criticality safety of the UF_6 handling area collection tanks is administrative control. The controls are implemented through administrative procedures which prohibit the discharge of liquids for cleaning radioactive spills into drains which feed the tanks. The applicant predicts that the uranium concentration will not exceed one ppm on the basis of Urenco experience. In addition, the tanks are sampled weekly.

The NRC staff has reviewed the applicant's administrative control program. The primary controls are a set of procedures to prohibit the intentional transfer of uranium solution to the collection tanks, and weekly sampling of the tank contents. The NRC staff notes that, although an unacceptable sample result may prompt actions to prohibit the future additions of uranium solution, it does not prohibit the addition of excessive uranium in the week following an acceptable weekly sample. The NRC staff also notes the claim by the applicant that, in over 30 years' operation of Urenco plants, "the typical contamination levels in these tanks are usually less than 1 ppm U." History, however, is not a sufficient basis for safety. In addition, the applicant did not evaluate the safety of the effluent pit for two particular events, that is, solution flowing into the pit from the process floor or the contents of a tank leaking onto the pit floor. On the other hand, the NRC staff considers the probability of a significant release of UF_6 followed by a large release of water, for example, during fire suppression, to be very low. This low probability results from the double-confinement design for processes involving large quantities of non-solid UF_6 and the low inventory of uranium in other processes. Even though the probability is very low, the need for a criticality monitor alarm system exists. The applicant will provide a system to monitor the TSA area.

To implement commitments in the SAR, the applicant has revised the PLC so that, at least weekly, the licensee will analyze two independent samples of the contents of each effluent collection tank in the UF_6 handling area. Agreement between sample results will be obtained, or the tank will be resampled. If the uranium content exceeds 100 ppm, the licensee will sample the tank on each operating shift until the source of the uranium is identified and controlled.

Effluent Collection Tanks, TSA

There are two 2,461-liter (650-gallons) tanks in the TSA effluent pit. The sources of liquid wastes include drains from the Chemistry Lab floor, Personnel Decontamination Area, Contaminated Equipment Workshop, Radioactive Waste Storage Area, other storage areas, Decontamination Workshop floor, and the Laundry floor. Flush water from the LWD dryer is also directed to these tanks. Approximately 2,840 liters (750 gallons) of liquids are generated each week. The largest source of uranium (20 gm/yr) is expected to be from the LWDS dryer. Each tank is sampled for uranium concentration when half full and before discharge. In the revised PLC, the applicant commits to analyze two independent samples of each TSA effluent collection tank when one-half full and before discharge. Agreement between results will be obtained, or the sampling process will be repeated until agreement is obtained. If the contents exceeds 1000 grams uranium, corrective action will include hourly sampling until the source of the uranium is identified and controlled.

Potential abnormal events identified by the applicant include erroneous discharges of sample solutions, Fomblin oil sludge solutions, or decontamination solutions, and containers of radioactive waste which leak into the TSA drains. The applicant's basis for safety is based in part on operating experiences in Urenco plants. In CSER Table 3, the applicant estimates that 378 kilograms of uranium are processed annually through the TSA. Of this quantity, 334 kilograms are transferred into the TSA and are repackaged and transferred as a solid. The applicant asserts that it is inconceivable that this material could enter TSA tanks as solution. The other 44 kilograms would be processed over a one-year period. Storage of this material, most of which would be in liquid form, is in favorable geometry containers inside a non-draining, curbed area. Uranium in sample solutions is limited to a safe mass. Fomblin oil sludge solutions are limited to safe volume containers. In addition, procedures are in place for the handling, storing, and treating these waste solutions. Inadvertent discharge of routinely generated process solutions should not lead to uranium quantities sufficient to form a critical mass. The applicant estimates that it would take 4 years and 23 weeks, respectively, to accumulate a mass sufficient to become critical from either Fomblin oil sludge or sample solutions.

The NRC staff has reviewed the applicant's safety analysis and agrees that, under normal conditions, a quantity of uranium equivalent to a critical mass would not be accumulated. The applicant does not identify necessary and sufficient controls to prevent the accumulation of more than a safe mass in the collection tanks, inside the curbed area, or in the tank pit area. On the other hand, neither the NRC staff nor the applicant identifies a failure mode whereby a critical configuration of enriched material can occur. The need for positive safety controls would further reduce the probability of a critical configuration, but pending identification of such a failure mode, additional controls cannot reasonably be required. The area is monitored by a criticality alarm system.

Citric Acid Baths and Spent Acid Collection Tank

In the Decontamination Facility, citric acid baths, one 1,703-liter (450-gallon) bath for large components such as pumps and one 378-liter (100-gallon) bath for smaller components such as valves are used for decontamination. All components to be decontaminated are inspected by the consigning department. Records transferred by the consigning department must indicate any loose material or blockage. The decontamination department then examines the document and re-inspects the component. Hidden surfaces are examined at this stage. If the component has gross contamination or blockage, the component is transferred to a ventilated workplace where the material or blockage is removed.

For nuclear criticality safety, the applicant proposes to limit the citric acid baths collectively to one safe mass. Control is exercised by estimating the uranium content of each basket of components before entry into the bath. In addition, samples are taken at least weekly and before discharge to the collection tank. Both baths have ultrasonic agitation, recirculating pumps, and spray nozzles to ensure representative sampling. The applicant proposes to use Urenco operating experience to log estimated additions to the two baths. The accumulation rate is expected to be less than several hundred grams of uranium per day, an amount which compares favorably with the 17-kilogram limit for a safe mass. The applicant uses concentration control, that is, 6.35 grams of uranium per liter to ensure that the two baths contain less than 17 kilograms of uranium. On the basis of logged values, a bath is sampled when the uranium concentration approaches 40 percent of the concentration limit. On the basis of logged values, the bath is sampled again at 70 percent of the concentration limit. The logged values are adjusted after both samples are analyzed. At 85 percent of the concentration limit, the bath is sampled and transferred to the collection tank. In the revised PLC, the applicant commits to analyze two independent samples of the citric acid baths weekly, as well as independent samples when tank contents reach estimated 40, 70, and 85 percent of the tank safe mass limit. Agreement will be obtained between both analyses, or new samples will be analyzed. If the estimated limits are exceeded, corrective actions will include adjusting the tank inventory record and investigating the cause of the nonconservative estimate.

There is one 2,461-liter (650-gallon) spent citric acid collection tank in the TSA effluent pit. For nuclear safety, the applicant proposes that the tank be sampled after each transfer. A log is maintained for each transfer although the concentration control on the baths should prevent the tank from accumulating a safe mass. The applicant calculates that the expected throughput in the tank is 33.7 kilograms of uranium per year.

The NRC staff believes that the use of Urenco data to estimated uranium inputs to the baths may not be appropriate for CEC operations. With the PLC commitment above, the NRC staff agrees that concentration control in the baths may be used to effect concentration control and, hence, mass control in the spent acid tank.

Decontamination Effluent Tank

The 2,461-liter (650-gallon) decontamination effluent tank in the TSA pit receives discharges from the two rinse baths in the Decontamination Facility and from clarified liquids from the LWDS tank. Before any transfer of liquids to the effluent tank, the applicant samples the liquids to ensure that not more than a safe mass is transferred. Subject to the license requirement for dual independent samples and analyses, the NRC staff agrees with the applicant's basis for safety.

In the Decontamination Facility, potentially contaminated water is generated in one 1,703-liter (450-gallon) bath and one 378-liter (100-gallon) rinse bath. Components decontaminated in the citric acid baths are washed in one of the rinse baths. The applicant's basis for safety of the rinse baths is that only contamination gets to the rinse tanks. Weekly sampling is sufficient to limit both baths to a single safe mass. The applicant estimates that the annual throughput is 12,491 liters (3,300 gallon) containing 840 grams of uranium.

The LWDS reaction tank is the second source of liquids to be transferred to the decontamination effluent tank. The reaction tank is sampled before transfer to ensure that not more than a safe mass of uranium is transferred at one time. The applicant estimates that the annual throughput is 11,792 liters (3,115 gallon) containing 36 grams of uranium. The LWDS process is discussed below.

With proper control on the citric acid decontamination step, the NRC staff agrees with the applicant's proposed sampling frequency to control the bath contents and to transfer the used rinse water to the decontamination effluent tank. With proper control, the combined annual throughput is less than a safe mass. With the applicant's commitment to dual sampling and analyses to establish mass control, the NRC staff agrees with the applicant's evaluation.

Laundry Tank

There is one 7,570.8-liter (2,000-gallon) tank in the TSA to collect laundry effluent. The effluent comes only from washing machines for laundering clothes used in the radiation control zone. All laundry is sorted and surveyed for radioactive contamination before washing. Grossly contaminated articles are discarded as solid waste. The waste water is expected to contain only traces of uranium contamination. Overall, the applicant estimates that less than 0.1 kilogram of uranium is processed through 689,000 liters (182,000 gallons) each year.

The applicant's basis for criticality safety is that significant quantities of uranium cannot be released onto clothing or cloths and go undetected into the laundry. Standard survey techniques would have to be totally ineffective for enough uranium to collect in any laundry batch or effluent collection tank. The NRC staff agrees that, with the clothing surveys, criticality is unlikely in the laundry effluent system.

Laboratory Tanks

The applicant estimates that the throughput in the laboratory is 3.5 kilograms of uranium per year. Most of this material is collected as waste in 5-liter bottles and transferred to the LWDS tank for processing. Another waste stream consisting of laboratory equipment washwater is collected in two 1,893-liter (500-gallon) tanks. Each tank is sampled before discharge.

The applicant's basis for criticality is administrative in nature. Because the throughput is less than one safe mass per year, no additional controls are necessary. The NRC staff agrees that a safe mass limit, specified below, for the laboratory is sufficient.

LWDS Reaction Process

In the LWDS reaction process, potassium hydroxide is used to precipitate uranium out of waste streams. A centrifuge removes the uranium precipitate from the waste stream. The three waste streams treated in the process are the spent citric acid, uranyl solutions from the laboratory, and uranyl solutions from the sample bottle cleaning process. The waste streams are precipitated in a 170-liter (45-gallon) tank and then recirculated through the centrifuge. When the solution has been clarified, the liquid and the citric cake are sampled.

The applicant estimates that approximately two safe masses will be treated in this system per year. Accordingly, the applicant proposes to use safe mass control for the process. After a safe mass has been processed, the tank and centrifuge internals are inspected for solids buildup. The NRC staff agrees that the proposed nuclear criticality control program is adequate.

LWDS Dryer Process and Effluent Monitor Tanks

All of the waste streams discussed above are processed through the LWDS dryer process. After satisfactory sample results are obtained, further processing is done without nuclear criticality safety considerations.

All of the waste streams discussed above are pumped through a filter into a 18,927-liter (5,000-gallon) LWDS collection tank. After feed adjustment, the waste liquids are pumped into the dryer where solids are collected in a 114-liter (30-gallon) drum, and the vapor condensate is collected in the distillate tank and then pumped into one of three 5,678-liter (1,500-gallon) effluent monitor tanks. When the monitor tank is full, it is sampled and either reworked or discharged to the sewage treatment system. If rework is necessary, the distillate can be processed through the dryer again or processed through the demineralizers.

The applicant's basis for nuclear safety is that all equipment for the dryer process is limited to not more than a safe mass. The applicant estimates that the annual throughput is 1,040,985 liters (275,000 gallons) of liquids, which will contain 1,820 kilograms of solids. The solids contain approximately 1 kilogram of uranium. The NRC staff agrees that mass control is adequate to effect nuclear safety. Because of the low inventory, no interaction analysis is necessary. The applicant's commitment to dual sampling and analyses to effect safe mass control is acceptable to the NRC staff.

9.3.12 Ventilation Systems

The applicant designed two CEC ventilation systems which handle air potentially contaminated with uranium, that is, the GEVS and the TSA HVAC system. The GEVS removes trace quantities of uranium and HF from contaminated or potentially contaminated process gas streams. The system is connected to the feed, tails, and product cylinder vent systems; autoclaves; and discharge lines from mobile vacuum pumps. Gases from the process gas streams pass through one of five parallel trains, each consisting of a pre-filter and a HEPA filter to remove UO_2F_2 particles. The TSA HVAC system serves contaminated or potentially contaminated areas not served by the process off-gas system. The air passes through one of three parallel HEPA filter plenums.

The applicant calculated that a maximum 3.0 kilograms of uranium accumulates on the GEVS and HVAC filters per year. This calculation is based on Urenco experience. The applicant asserts that this deposition rate, together with regular sampling of the filters, does not constitute a potential for criticality. In the revised PLC, the applicant commits to investigations to evaluate ductwork deposition at least every three years (maximum interval 42 months).

At least every three years, LES will survey the ventilation ducts for uranium deposition. If uranium deposition (other than surface contamination) is found, corrective action must include removal of uranium. Alternatively, LES may choose to demonstrate with measurements that the total quantity of U-235 in a ductwork system is less than half of the safe critical mass, based on safe critical mass values specified in PLC Table 4.2-1. For this latter approach, U-235 in any connected ductwork which could conceivably combine in normal or abnormal operating conditions, must be added to determine the total U-235 mass of the system. Corrective action will be to remove U-235 such that the total is less than half of the safe critical mass. The NRC staff finds this approach acceptable.

9.3.13 Degreaser Baths

The degreaser baths remove oil substances from pump components. The two baths for this operation each contain 56.8 liters (15 gallons) of degreaser solvent.

For criticality control purposes, the applicant proposes to weigh each basket of components before and after degreasing. The weight loss is assumed to be uranium sludge in the degreaser. When the weight loss equals the safe mass weight for uranium, the sludge is removed from the degreaser. The NRC staff agrees that this control is adequate.

9.3.14 Chemical Lab

The applicant provides no explicit safety analysis for the Chemistry Laboratory. In discussing the nuclear safety controls for laboratory wastes, however, the applicant states that 3.5 kilograms of uranium is processed through the laboratory each year. To establish a nuclear safety limit, the applicant has made commitments in the revised PLC to limit the Chemical Laboratory area to not more than 10.0 kilograms of uranium. The administrative limit will be ensured by maintaining a current inventory by logging transfers into and out of the laboratory.

9.3.15 Staff's Evaluation of Nuclear Criticality Safety Criteria in Plant Design and Operation

The NRC staff has evaluated the applicant's proposed plant equipment design and operation. The applicant establishes safety limits and controls for each major step in the process. The limits are established with due consideration of the double-contingency principle. Subject to NRC staff's verification that the plant is built as described above and in the referenced applicant documents, the NRC staff finds the facility can be operated safely with respect to nuclear criticality concerns.

The applicant requested an exemption from the requirement for a criticality monitoring and alarm system for part of the facility. The NRC staff agrees that criticality is extremely unlikely in the cascade halls, for example. However, in other areas, such as the TSA, the NRC staff believes that criticality is not as unlikely. The applicant does not discuss other areas of the facility such as the outside storage area and the shipping building. Because the applicant will install a monitor/alarm system, the NRC staff believes that detectors should be installed in all enriched uranium handling and storage areas. In the revised SAR, the applicant agreed to install a criticality monitor alarm system as required by Section 70.24 of Title 10 of the Code of Federal Regulations (CFR).

9.4 Staff Conclusions on Applicant's Nuclear Criticality Safety Program

The NRC staff has carefully considered the descriptive material and proposed commitments in the revised SAR, the revised CSER, and the revised PLC with respect to nuclear criticality safety. The proposed CEC organization, administrative practices for nuclear criticality safety, and the technical safety criteria for design and operation of the CEC facility provide reasonable assurance that the applicant can possess source material for the purpose of producing SNM without any undue risk to the health and safety of the public and workers or to the environment from nuclear criticality events. The NRC staff recommends that, with respect to nuclear criticality safety, the applicant be granted a license to construct and operate the CEC. This recommendation for operation is subject to NRC staff's verification that the CEC has been built and tested in accordance with the commitments described in the above referenced LES documents.

10 CONDUCT OF OPERATIONS

10.1 Background

An applicant for a license under Parts 40 (NRC, 1961) and 70 (NRC, 1956a) of Title 10 of the Code of Federal Regulations (CFR) must provide information concerning the control or ownership of the applicant; the organizational structure and technical qualifications of the applicant, including the training and experience of the applicant's staff; and proposed procedures including management controls. Additionally, no licensee may commence operation of a uranium enrichment facility until the NRC verifies through inspection that the facility has been constructed in accordance with the requirements of the license.

The NRC staff has reviewed and evaluated the applicant's proposed organization, technical qualifications, procedures and management controls, as required by 10 CFR Parts 40 and 70, to determine if they are adequate to protect health and minimize danger to life and property.

The NRC staff's findings are discussed in each of the appropriate sections of this report. Section 10.8 addresses Emergency Planning. Section 10.9 addresses the inspection to be conducted by the NRC staff before commencement of enrichment operations. Section 10.10 is the NRC staff's conclusion.

10.2 Organization

Louisiana Energy Services (LES), the owner and operator of the Claiborne Enrichment Center (CEC), is a Delaware-registered limited partnership formed to provide uranium enrichment services for commercial nuclear power plants as its only business. LES has no subsidiaries or divisions.

The four LES general partners are:

- Claiborne Energy Services, Inc. (a Louisiana corporation and wholly-owned subsidiary of Duke Power Company, a publicly-held North Carolina corporation)

- Claiborne Fuels L.P., a Delaware-registered limited partnership of which Claiborne Fuels, Inc., a California corporation and wholly-owned subsidiary of Fluor Daniel, Inc. (FDI), a California corporation and wholly-owned subsidiary of Fluor Corporation, a publicly held Delaware-registered corporation, is the sole partner

- Graystone Corporation (Graystone), a Minnesota corporation and a wholly-owned subsidiary of Northern States Power Company, a publicly-held Minnesota corporation

- Urenco Investments, Inc. (Urenco), a Delaware-registered corporation and wholly-owned subsidiary of Urenco, Ltd., a corporation formed under the laws of the United Kingdom and owned in equal shares by International Nuclear Fuels plc (INFL), a public limited company formed under English law ; Ultra-Centrifuge Netherlands NV (UCN), a corporation formed under Netherlands laws; and Uranit GmbH (Uranit), a corporation formed under the laws of the Federal Republic of Germany. INFL is wholly-owned by the Government of the United Kingdom. UCN is owned by the Government of the Netherlands (99 percent) and by the Royal Dutch Shell Group, the Dutch State Mines, Philips Gloeilampenfabrieken N.V. and VMF-STORK (1 percent, collectively). Uranit is owned by PreussenElektra AG (37.5 percent), RWE AG (37.5 percent) and Hoechst AG (25 percent), all of which are corporations formed under laws of the Federal Republic of Germany.

The seven limited partners are as follows:

- BNFL Enrichment (Investments US) Ltd., a corporation formed under the laws of the United Kingdom and a wholly owned subsidiary of BNFL

- Claiborne Energy Services, Inc. (see above)

- GnV, a corporation formed under the laws of the Federal Republic of Germany and a wholly owned subsidiary of Uranit

- Le Paz Inc., a Minnesota corporation and wholly owned subsidiary of Graystone

- Louisiana Power & Light Company, a Louisiana corporation and wholly owned subsidiary of Entergy Corporation, a publicly held Florida corporation and a public utility holding company

- Micogen Limited III, Inc., a California corporation and wholly owned subsidiary of FDI

- UCN Deelnemingen B.V., a Netherlands corporation and wholly owned subsidiary of UCN.

LES is responsible for the design, quality assurance, construction, preoperational testing, initial start-up, and operation of the facility. The President of LES reports to the LES Management Committee. This Committee is composed of representatives from the four general partners of LES. Figure 10.1 outlines this relationship.

The operating organization for LES is shown in Figures 10.2 and 10.3. The positions shown are functional and may not correspond to actual titles. As CEC construction nears completion, LES will staff the facility at sufficient levels to ensure a smooth transition from construction to operation activities through training, procedure development, and other pre-operational activities. Urenco personnel are integrated into the organization to provide technical support during initial start-up of the facility.

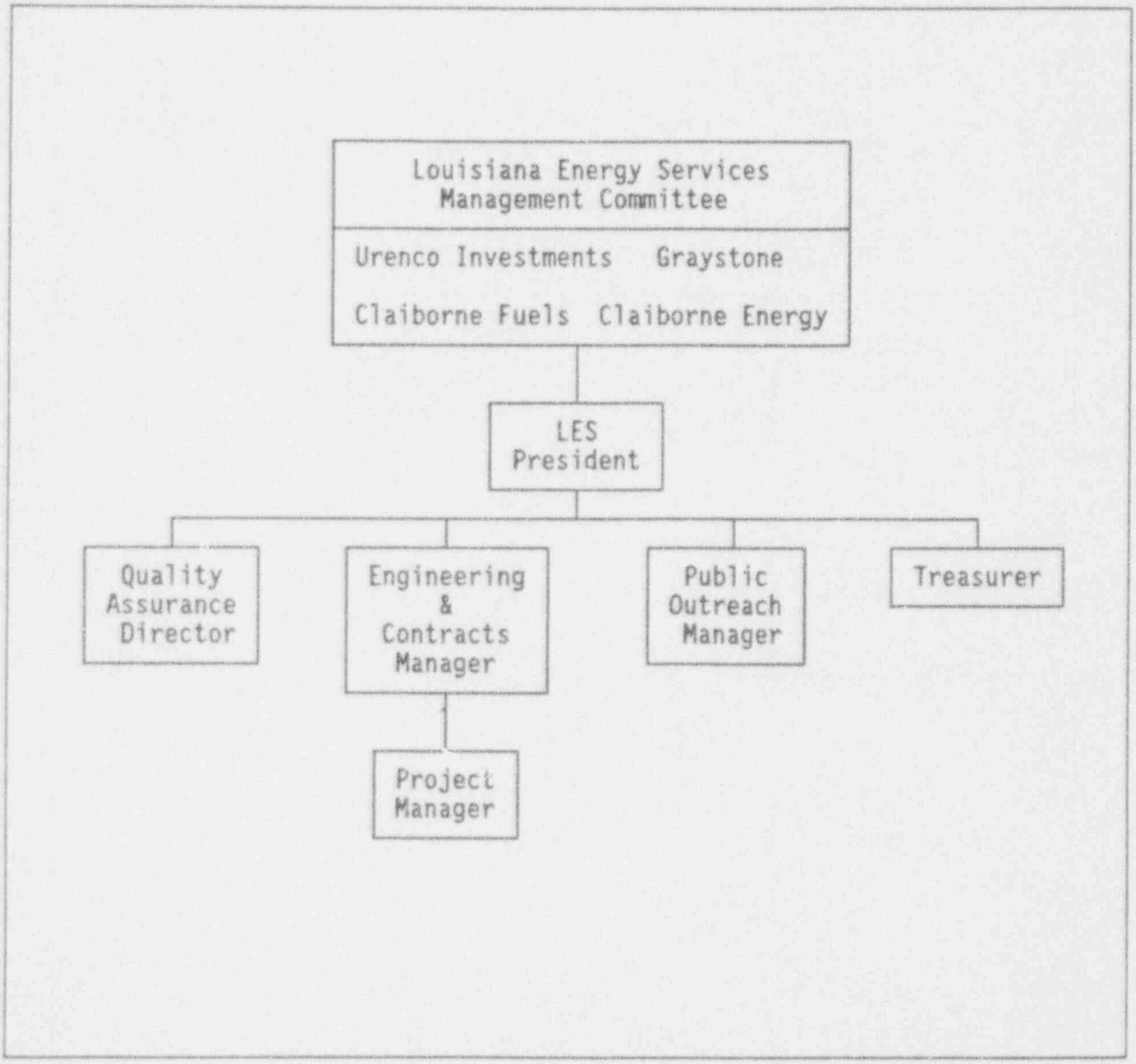

Figure 10.1 Louisiana Energy Services organization

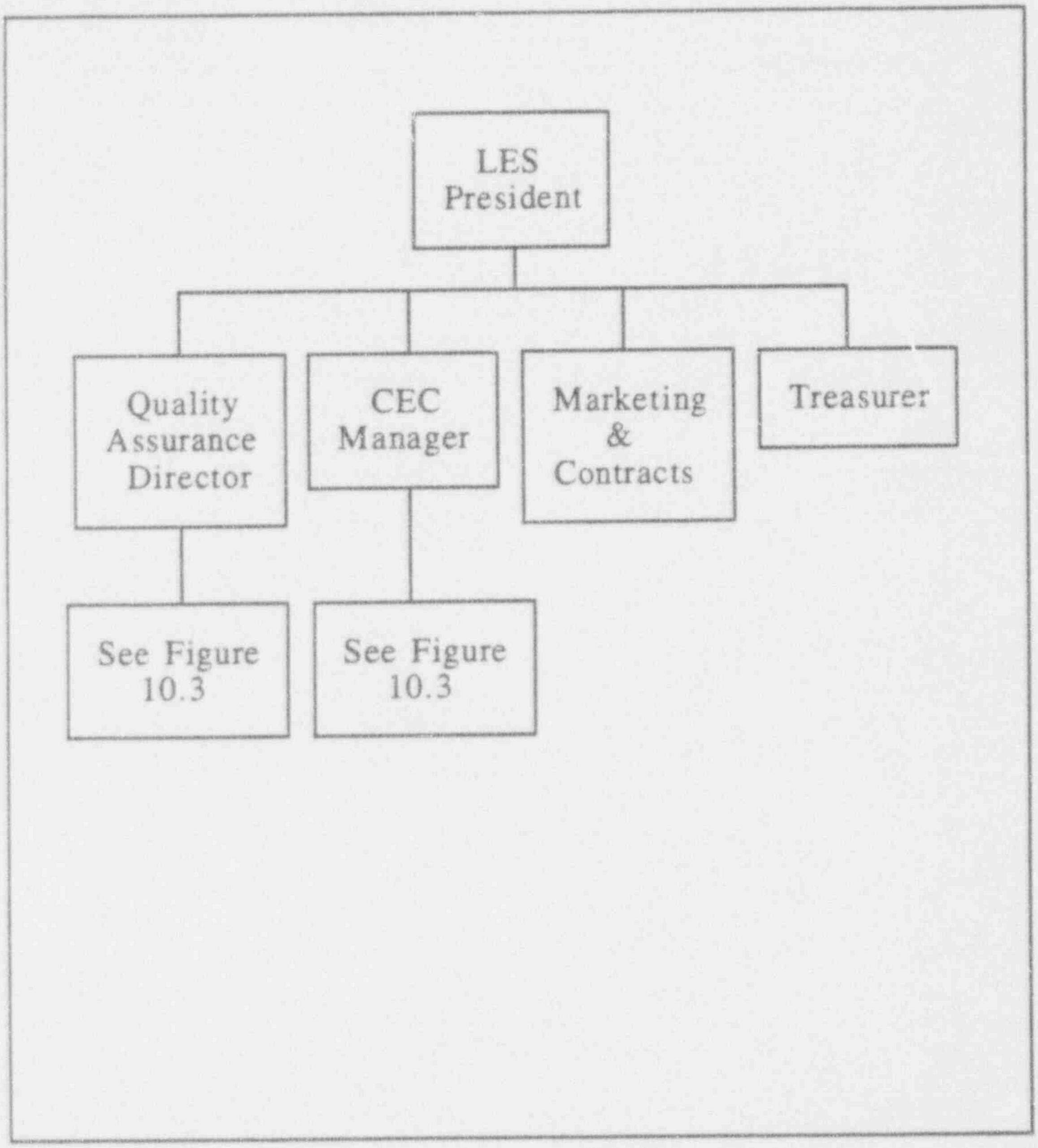

Figure 10.2 Louisiana Energy Services construction and operating organization

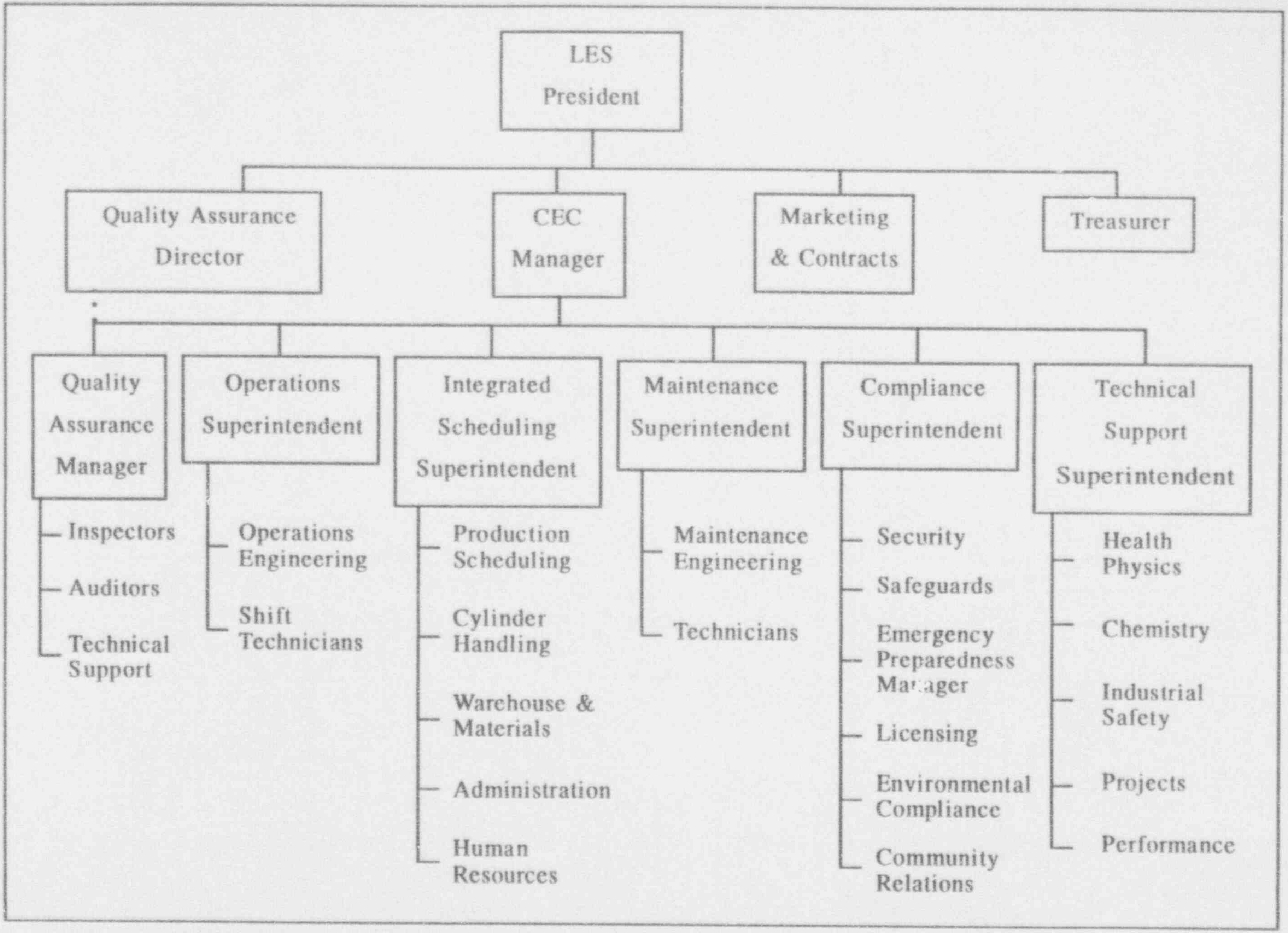

Figure 10.3 Louisiana Energy Services Claiborne Enrichment Center operating organization

LES has contracted Urenco to prepare the reference design for the facility, and with FDI and Duke Engineering & Services, Inc., to further specify structures and systems of the facility, as well as materials and equipment. Urenco has experience in the gas centrifuge uranium enrichment process because it operates three gas centrifuge uranium enrichment sites in Europe.

Preparation of construction documents and construction itself is contracted to qualified contractors. Urenco will design, manufacture and deliver to the site the centrifuges necessary for facility operation. In addition, Urenco is supplying technical assistance and consultation for the facility. Urenco is conducting technical audits of the design activities of FDI and Duke Engineering & Services to ensure that the CEC design is in accordance with the Urenco reference design.

10.3 Staff Qualifications, Functions, and Responsibilities

The minimum required qualifications, functions, and responsibilities of supervisory staff directly involved with safe CEC operation are outlined below.

The minimum qualification requirements for the facility functions that are directly responsible for its safe operation shall be as outlined below. The nuclear experience of each individual shall be determined to be acceptable by the CEC Manager. "Responsible nuclear experience" for these positions shall include (a) responsibility for and contributions towards support of facility(s) in the nuclear fuel cycle (for example, design, construction, and/or decommissioning), and (b) experience with chemical materials and/or processes. Different experience requirements may be approved by the CEC Manager only as specified in the following requirements for key positions. This shall be done in writing and only on a case-by-case basis.

The minimum qualifications for the following Managers and Superintendents are specified in the applicant's Proposed License Conditions (PLCs) (LES, 1993e):

- CEC Manager
- Quality Assurance Manager
- Operations Superintendent
- Integrated Scheduling Superintendent
- Maintenance Superintendent
- Compliance Superintendent
- Technical Support Superintendent
- Security Manager
- Safeguards Manager
- Emergency Preparedness Manager
- Health Physics Manager

- Projects Manager
- Chemistry Manager
- Industrial Safety Manager

CEC Manager

The CEC Manager shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and six years of responsible nuclear experience.

The CEC Manager shall be appointed by, and report to, the President of LES. The CEC Manager has direct responsibility for operation of the facility in a safe, reliable, and efficient manner. The CEC Manager shall be knowledgeable of the enrichment process, enrichment process controls and ancillary processes, criticality safety control, chemical safety, industrial safety, and radiation protection program concepts as they apply to the overall safety of a nuclear facility. The CEC Manager is responsible for appropriate selection of CEC staff for all key positions, including membership of the Facility Safety Review Committee (FSRC).

The CEC Manager will be responsible for the protection of the facility staff and the general public from radiation and chemical exposure and/or any other consequences of an accident at the facility and will also bear the responsibility for compliance with the facility license. The CEC Manager or designee(s) will have the authority to approve and issue Department Directives and procedures.

Quality Assurance Manager

The Quality Assurance (QA) Manager shall have a BS degree (or equivalent) and minimum of four years of responsible nuclear experience in the implementation of a quality assurance program. To be acceptable this academic training shall be in engineering or scientific fields. The QA Manager shall have at least two years experience in a QA organization at a nuclear facility.

The QA Manager will report to the CEC Manager and will be responsible for implementing the QA Program for the facility. This includes responsibility for ensuring that all activities at the facility affecting quality are being performed in accordance with appropriate regulations, codes and standards. This position will be independent from other management positions at the facility to ensure that the QA Manager has direct access to the CEC Manager for matters affecting quality. In addition, the QA Manager will have the authority and responsibility to contact directly the QA Director and/or the President of LES with any quality assurance concerns.

Operations Superintendent

The Operations Superintendent shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

The Operations Superintendent will report to the CEC Manager and will have the responsibility of directing the day-to-day operation of the facility. This includes such activities as ensuring the correct and safe operation of the UF_6 processes, proper handling of UF_6, and the periodic testing of equipment to ensure safe and efficient operation. In the event of the absence of the CEC Manager, the Operations Superintendent may assume the responsibilities and authorities of the CEC Manager.

Integrated Scheduling Superintendent

The Integrated Scheduling (IS) Superintendent shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

The IS Superintendent will report to the CEC Manager and will have the responsibility of directing the scheduling of enrichment operations to ensure smooth production at the facility. This includes such activities as ensuring proper feed material and maintenance equipment is available for the facility. The IS Superintendent will also be responsible for providing administrative and human resource services to the facility, including the training program. In the event of the absence of the CEC Manager, the IS Superintendent may assume the responsibilities and authorities of the CEC Manager.

Maintenance Superintendent

The Maintenance Superintendent shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

The Maintenance Superintendent will report to the CEC Manager and will have the responsibility of directing and scheduling maintenance activities to ensure proper operation of the facility. This includes activities such as repair and preventive maintenance of facility equipment. In the event of the absence of the CEC Manager, the Maintenance Superintendent may assume the responsibilities and authorities of the CEC Manager.

Compliance Superintendent

The Compliance Superintendent shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

The Compliance Superintendent will report to the CEC Manager and will have the responsibility of directing the activities that ensure that the facility maintains compliance with appropriate regulations and conformance with applicable codes. This includes activities associated with physical security, classified matter and information, licensing, emergency preparedness, safeguarding of special nuclear material and compliance with environmental regulations. The Compliance Superintendent or designee will review and approve changes to the CEC facility or operations which involve a change to the facility as described in the SAR. In the event of the absence of the CEC Manager, the Compliance Superintendent may assume the responsibilities and authorities of the CEC Manager.

Technical Support Superintendent

The Technical Support (TS) Superintendent shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and four years of appropriate, responsible nuclear experience.

The TS Superintendent will report to the CEC Manager and will have the responsibility of providing technical support to the facility. This includes activities associated with health physics, chemistry, industrial safety and engineering including criticality safety reviews, and computer support. The TS Superintendent or designee will review and approve changes to the facility or to operations which involve chemical, radiation, or criticality hazards. In the event of the absence of the CEC Manager, the TS Superintendent may assume the responsibilities and authorities of the CEC Manager.

Security Manager

The Security Manager shall have a minimum of five years of experience in the responsible management of physical security at a facility requiring security capabilities similar to those required for the CEC. No credit for academic training may be taken toward fulfilling this experience requirement.

The Security Manager will report to the Compliance Superintendent and will have the responsibility for directing the activities of security personnel to ensure the physical protection of the facility. The Security Manager will also be responsible for the protection of classified matter and information at the facility and obtaining proper security clearances for facility personnel and support personnel. In matters involving physical protection of the facility or classified matter, the Security Manager will have direct access to the CEC Manager.

Safeguards Manager

The Safeguards Manager shall have a minimum of five years of experience in the management of a safeguards program for special nuclear material, including responsibilities for material control, material accountability, and physical security. No credit for academic training may be taken toward fulfilling this experience requirement.

The Safeguards Manager will report to the Compliance Superintendent and will have the responsibility for ensuring the proper implementation of the Fundamental Nuclear Material Control (FNMC) Plan. This position will be separate from and independent of the operations, maintenance, and technical support departments to ensure a definite division between the safeguards group and the other departments. In matters involving safeguards, the Safeguards Manager will have direct access to the CEC Manager.

Emergency Preparedness Manager

The Emergency Preparedness Manager shall have two years of experience in the implementation of emergency plans and procedures at a nuclear facility. No credit for academic training may be taken toward fulfilling this experience requirement.

The Emergency Preparedness Manager will report to the Compliance Superintendent and will have the responsibility for ensuring the facility can react and respond to any emergency situation that may arise. This includes emergency preparedness training of all facility and facility support personnel and conducting periodic drills to ensure that personnel training is up-to-date. The plans for maintaining the capability to respond to emergency situations at the CEC are detailed in the Louisiana Energy Services CEC Emergency Plan.

Health Physics Manager

The Health Physics Manager shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and three years of responsible nuclear experience associated with implementation of a health physics program. At least two years of experience shall be at a facility that processes uranium, including uranium in soluble form.

The Health Physics Manager will report to the Technical Support Superintendent and will have the responsibility for implementing the health physics program. These duties include the training of personnel in use of radiological program support equipment, control of radiation exposure of personnel, continuous determination of the radiological status of the facility, determining the need for, issuing of, and closing out of radiation work permits, and conducting the radiological environmental monitoring program. In matters involving radiological protection, the Health Physics Manager will have direct access to the CEC Manager.

During emergency conditions the Health Physics Manager's duties will include:

- providing Emergency Operations Center personnel information and recommendations concerning chemical and radiation levels at the facility
- gathering and compiling onsite and offsite radiological and chemical monitoring data
- making recommendations concerning actions at the facility and offsite deemed necessary for limiting exposures to facility personnel and members of the general public, and
- taking prime responsibility for decontamination activities.

Projects Manager

The Projects Manager shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and three years of appropriate, responsible nuclear experience. The Projects Manager shall also have at least one year of direct experience in the administration of nuclear criticality safety reviews.

Within the Projects group shall be a Projects Analyst with a minimum of two years experience in the implementation of a criticality safety program. This individual shall hold a BS degree in an engineering or scientific field and have successfully completed a training program, appropriate to the scope of operations, in the physics of criticality and in associated safety practices.

Should a change to the facility require a nuclear criticality safety evaluation, the analysis shall be performed by an individual who, as a minimum, possesses the qualifications of the Projects Analyst. An independent review of the analysis shall be performed by an individual who, as a minimum, has the education and training of a Projects Analyst. In addition, this individual shall have at least two years of experience performing criticality safety analysis and implementing criticality safety programs.

The Projects Manager will report to the Technical Support Superintendent and will have the responsibility for the implementation of facility modifications and the approval of facility procedures and modifications for criticality safety. The Projects Manager will also provide engineering support as needed to support facility operation and maintenance. This support will include performance testing of systems and equipment.

Chemistry Manager

The Chemistry Manager shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and three years of appropriate, responsible nuclear experience associated with implementation of a facility chemistry program.

The Chemistry Manager will report to the Technical Support Superintendent and will have the responsibility for the implementation of chemistry and safety programs and procedures for the facility. This includes chemical analysis of facility effluents and environmental samples, chemical safety programs, and reporting effluent chemical analyses to regulatory agencies.

Industrial Safety Manager

The Industrial Safety Manager shall have, as a minimum, a BS degree (or equivalent) in an engineering or scientific field and three years of appropriate, responsible nuclear experience associated with implementation of a facility safety program.

The Industrial Safety Manager will report to the Technical Support Superintendent and will have responsibility for implementation of facility industrial safety programs and procedures. This shall include programs and procedures for training individuals in safety and maintaining performance of the facility fire protection system.

Performance Manager

The Performance Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear performance program.

The Performance Manager will report to the Technical Support Superintendent and will have the responsibility for coordinating and maintaining testing programs for the facility. This includes testing of systems and components to ensure the systems and components are functioning as specified in design documents.

Licensing Manager

The Licensing Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear licensing program.

The Licensing Manager will report to the Compliance Superintendent and will have the responsibility for coordinating facility activities to ensure compliance is maintained with applicable NRC requirements. The Licensing Manager will also be responsible for ensuring abnormal events are reported to the NRC in accordance with NRC regulations.

Environmental Compliance Manager

The Environmental Compliance Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear environmental compliance program.

The Environmental Compliance Manager will report to the Compliance Superintendent and will have the responsibility for coordinating facility activities to ensure compliance with all local, state and federal environmental regulations. This includes submission of periodic reports to appropriate regulating organizations of effluents from the CEC.

Community Relations Manager

The Community Relations Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a community relations program.

The Community Relations Manager will report to the Compliance Superintendent and will have the responsibility for providing information about the CEC and LES to the public and media. During an abnormal event at the facility, the Community Relations Manager will ensure that the public and media receive accurate and up-to-date information.

Maintenance Engineering Manager

The Maintenance Engineering Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear maintenance program.

The Maintenance Engineering Manager will report to the Maintenance Superintendent and will have the responsibility for providing maintenance support for equipment and systems at the CEC. This includes preventive maintenance of appropriate equipment to ensure systems and equipment at the facility operate safely and as designed.

Maintenance Technicians Manager

The Maintenance Technicians Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear maintenance program.

The Maintenance Technicians Manager will report to the Maintenance Superintendent and will have the responsibility for providing maintenance support for equipment and systems at the CEC. This includes periodic inspection and adjustment of equipment and systems at the CEC to ensure systems and equipment at the facility operate safely and as designed.

Production Scheduling Manager

The Production Scheduling Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a continuous production scheduling program.

The Production Scheduling Manager will report to the Integrated Scheduling Superintendent and will have the responsibility for developing and maintaining production schedules for enrichment services.

Cylinder Handling Manager

The Cylinder Handling Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a continuous production scheduling program.

The Cylinder Handling Manager will report to the Integrated Scheduling Superintendent and will have the responsibility for ensuring that cylinders of uranium hexafluoride are received and dispatched correctly at the CEC.

Warehouse and Materials Manager

The Warehouse and Materials Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a purchasing and inventory program.

The Warehouse and Materials Manager will report to the Integrated Scheduling Superintendent and will have the responsibility for ensuring spare parts and other materials needed for operation of the CEC are ordered, received, inspected and stored properly.

Administration Manager

The Administrative Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising administrative responsibilities at an industrial facility.

The Administration Manager will report to the Integrated Scheduling Superintendent and will have the responsibility for ensuring support functions such as accounting, word processing and document control are provided for the CEC.

Human Resource Manager

The Human Resource Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising human resource responsibilities at an industrial facility.

The Human Resource Manager will report to the Integrated Scheduling Superintendent. The Human Resources Manager will have responsibility for ensuring that adequate staffing and employee training is provided for the CEC.

Operations Engineering Manager

The Operations Engineering Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear operations program.

The Operations Engineering Manager will report to the Operations Superintendent and will have the responsibility for ensuring safe operation of enrichment equipment and support equipment. This includes the development of operating procedures for the CEC.

Operations Shift Technicians Manager

The Operations Shift Technicians Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear operations program.

The Operations Shift Technicians Manager will report to the Operations Superintendent and will have the responsibility for ensuring safe operation of enrichment equipment and support equipment. The Operations Shift Technicians Manager will direct personnel in order to provide enrichment services in a safe and efficient manner.

Quality Assurance Inspectors Supervisor

The Quality Assurance Inspectors Supervisor shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear quality assurance program.

The Quality Assurance Inspectors Supervisor will report to the QA Manager and will have the responsibility for performing inspections related to the implementation of the LES Quality Assurance Program.

Quality Assurance Auditors Supervisor

The Quality Assurance Auditors Supervisor shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear quality assurance program.

The Quality Assurance Auditors Supervisor will report to the QA Manager and will have the responsibility for performing audits related to the implementation of the LES Quality Assurance Program.

Quality Assurance Technical Support Supervisor

The Quality Assurance Technical Support Supervisor shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear quality assurance program.

The Quality Assurance Technical Support Supervisor will report to the QA Manager and will have the responsibility for providing technical support related to the implementation of the LES Quality Assurance Program.

The NRC staff finds that the applicant's organization and personnel qualifications are adequate for the proposed licensed activities. The NRC staff agrees with the applicants's commitments, by license conditions (LES, 1993e), for minimum educational requirements, appropriate responsible nuclear experience, and responsibilities for managers and superintendents in key safety related positions.

10.4 Training

By the applicant's PLCs (LES, 1993e), formal, planned training programs shall be established for CEC employees. Indoctrination training shall be provided to all employees before the employees perform work at the CEC and shall address:

- safety preparedness for all safety disciplines (i.e., criticality, radiological, chemical, industrial - including fire protection safety
- ALARA practices
- issues related to 10 CFR Part 19
- environmental protection
- emergency procedures.

In depth training programs shall be provided to individuals depending on job requirements in the areas of radiological safety (for all personnel with access to restricted areas) and in criticality safety control. Nuclear criticality safety training shall satisfy the requirements of ANSI/ANS 8.20 - 1991 "Nuclear Criticality Safety Training". All LES employees engaged in QA related activities shall receive performance based QA training. All visitors and contractors shall receive appropriate training prior to visiting the facility and/or performing work at the facility. Retraining of personnel previously trained shall be performed for radiological, chemical, industrial, and criticality safety at least annually, and shall include procedure changes and updates and changes in required skills. The training program shall include methods of verifying training effectiveness, such as written tests, actual demonstration of skills, and where required by regulation, maintaining a current and valid license demonstrating qualification. Changes to training shall be implemented if indicated due to incidents potentially compromising safety, or if changes are made to facilities or processes. Records of training successfully completed shall be maintained in accordance with Section 2.9 of the PLCs (LES, 1993e) for all personnel.

LES' training program will be designed to prepare initial and replacement personnel for the safe operation of the facility. The level at which an employee initially enters the training program will be determined by an evaluation of the employee's past experience, level of ability, and qualifications.

Facility personnel may be trained through participation in prescribed parts of the training program which will consist of the following:

- General Employee Training
- Technical Training
- Employee Development/Management-Supervisory Training

Training requirements shall be applicable to, but not necessarily restricted to, those personnel within the plant organization who have a direct relationship to the operation, maintenance or other technical aspects of the CEC. Training courses will be kept up-to-date to reflect plant modifications and changes to procedures.

Periodic retraining will be conducted to ensure retention of knowledge and skills important to safety.

Training programs shall be established for the various types of job functions (e.g., production operator, radiation protection technician, contractor personnel) commensurate with criticality radiation safety responsibilities associated with each such position. Visitors to the Controlled Access Area (CAA) will be trained in the formal training program or will be escorted by trained personnel while in the CAA.

10.4.1 General Employee Training

General Employee Training (GET) will encompass those quality assurance, radiation protection, safety, emergency and administrative procedures established by CEC management and applicable regulations. Continuing training will be conducted in these areas. All persons under the supervision of facility management must participate in GET; however, certain facility support personnel, depending on their normal work assignment, may not participate in all topics of the GET. Temporary maintenance and service personnel will receive GET to the extent necessary to assure safe execution of their duties. Certain portions of GET may be included in a New Employee Orientation Program.

GET topics are listed below.

- General administrative controls and procedure use
- Quality assurance policies and procedures
- Facility systems and equipment
- Nuclear safety (See below - includes the use of dosimetry, protective clothing, and equipment)
- Industrial safety, health, and first aid
- Emergency plan and implementing procedures
- Facility security programs (includes the protection of classified matter and information)
- Chemical Safety

- Fire protection and fire brigade (See below)

- New employee orientation

10.4.2 Nuclear Safety Training

This training will stress radiological and chemical safety of plant personnel and the public.

Personnel access procedures will ensure the completion of formal nuclear safety training prior to permitting unescorted access into the CAA.

Training sessions covering criticality safety, radiation protection and emergency procedures will be conducted on a regular basis to accommodate new employees or those requiring retraining. Topics covered in the training program will include:

- Notices, reports and instructions to workers

- Practices designed to keep radiation exposures ALARA

- Methods of controlling radiation exposures

- Contamination control methods

- Use of monitoring equipment

- Emergency procedures and actions

- Nature and sources of radiation

- Safe use of chemicals

- Biological effects of radiation

- Use of personnel monitoring devices

- Principles of nuclear criticality safety

- Risk to pregnant females

Individuals attending these sessions must pass an initial examination covering the training contents to assure the understanding and effectiveness of the training. The effectiveness of the training programs is also evaluated by audits of operational areas personnel responsible for criticality safety and health physics.

Newly hired or transferred employees reporting for work prior to the next regularly scheduled training session must complete formal nuclear safety training prior to unescorted access into the CAA. Nuclear safety training topics will include:

- Radiation protection practices
- Exposure monitoring devices
- Protective clothing
- Respiratory protection
- Personnel surveys
- Criticality safety
- Emergency actions

Since contractor employees will perform diverse tasks in the controlled area, formal training for these employees will be designed to address the type of work they perform. In addition to applicable radiation safety topics, training contents may include Radiation Work Permits, special bioassay sampling, and special precautions for welding, cutting, and grinding in the CAA.

These training programs will be conducted by instructors certified by the managers responsible for criticality safety and radiation protection. Records of the training programs are maintained as described in SER Section 10.4.6.

Individuals requiring unescorted access to the CAA will receive annual retraining. Retraining for individuals will be scheduled and reported by means of a computerized tracking system.

Contents of the formal nuclear safety training programs will be reviewed and updated as required at least every two years by the Technical Support Superintendent and Compliance Superintendent to ensure that the programs are current and adequate.

Operational personnel will be further instructed in the specific safety requirements of their work assignments by their immediate supervisor (or delegate) during on-the-job training. Employees must demonstrate understanding of work assignment requirements based on observations by their immediate supervisor (or delegate) before working without direct supervision.

Changes to work procedures including safety requirements, will be reviewed with operational personnel by their immediate supervisor or delegate.

Radiation and chemical safety topics will be discussed and reviewed at least annually in roundtable safety meetings held by supervisors or delegates with their workers, and at other meetings held by managers with their employees.

10.4.3 Fire Brigade Training

The primary purpose of the Fire Brigade Training Program will be to develop a group of facility employees skilled in fire prevention, fire fighting techniques, first aid procedures, and emergency response. They will be trained and equipped to function as a team for the fighting of fires. The intent of the facility fire brigade will be to be a first response effort designed to supplement the local fire department for fires at the plant and not to replace local fire fighters.

The Fire Brigade Training program will provide for initial training of all new fire brigade members, semi-annual classroom training and drills, annual practical training, and leadership training for fire brigade leaders.

10.4.4 Technical Training

Technical training will be designed, developed and implemented to assist facility employees in gaining an understanding of applicable fundamentals, procedures, and practices common to a gas centrifuge uranium enrichment facility. Also, technical training will be used to develop manipulative skills necessary to perform assigned work in a competent manner. Technical training will consist of four segments:

- Initial Training
- On-the-Job Training and Qualifications
- Continuing Training
- Special Training

10.4.4.1 Initial Training

Initial job training will be designed to provide an understanding of the fundamentals, basic principles, and procedures involved in work to which an employee is assigned. This training may consist of but will not be limited to, live lectures, taped and filmed lectures, self-guided study, demonstrations, laboratories and workshops exercises, and on-the-job training.

Certain new employees or employees transferred from other sections within the facility may be partially qualified by reason of previous applicable training or experience. The extent of further training for these employees will be determined by applicable regulations, performance in review sessions, comprehensive examinations, or other techniques designed to identify the employee's present level of ability.

Initial job training and qualification programs will be developed for operations, maintenance and technical services classifications. Training for each program will be grouped into logical blocks or modules and presented in such a manner that specific behavioral objectives are accomplished. Trainee progress will be evaluated using written examinations, oral or practical tests. Depending upon the regulatory requirements or individual's needs and plant operating conditions, allowances will be made to suit specific situations. A brief description of modules that may be contained in the initial Training Programs are as follows:

(1) Operations Initial Training

(a) General Systems

This training module will provide the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components will be taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment plant.

(b) Specific Systems

This training module will provide basic instruction in system and component identification and basic system operating characteristics. It will provide a general overview of enrichment plant equipment and acquaint the trainees with enrichment plant terminology and nomenclature and provide instructions describing basic system operations.

(c) Nuclear Preparatory

This training module will develop the necessary concepts in basic nuclear physics, plant chemistry, basic thermodynamics, radiation protection, and enrichment theory. Experience in enrichment control and radiation protection will also be provided. It will normally be presented to operations personnel following the Systems Specific Training Module.

(2) Mechanical Maintenance Initial Training

(a) Enrichment Plant Fundamentals

The Enrichment Plant Fundamentals Module will provide new employees with required material on administration, mathematics, physical science, systems, and safety.

(b) Fundamental Shop Skills

This training module will provide instruction in fundamentals of mechanical maintenance performance. It will combine academic instruction with hands-on training to familiarize trainees with design operational and physical characteristics of enrichment plant components, and basic skills and procedures used to perform mechanical repairs and/or equipment replacement. Task training lists will be integrated into this module to assure that each trainee attains a minimum level of performance. Tasks will be assigned and trainees will use work procedures to guide them through a task. Radiological, chemical and industrial safety will be stressed in all phases of this training module.

(c) Plant Familiarization

The Plant Familiarization module will provide for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the facility.

(3) Instrumentation, Electrical, and Performance Initial Training

(a) Enrichment Plant Fundamentals

The Enrichment Plant Fundamentals Module will provide new employees with required material on administration, mathematics, physical science, systems, and safety.

(b) Basic Instrument and Electrical

The Instrument and Electrical module will provide the trainee with refresher training in Electrical and Electronic Fundamentals, Digital Techniques and Application, Instrumentation and Control Theory and Application, and an introduction to the types and proper use of measuring and test equipment commonly used in enrichment plants.

The module will also provide the student a working knowledge of nuclear and non-nuclear instrumentation systems, overall integrated plant operation and control, and, in particular, the hazards of calibration errors and calibration during plant operation.

(c) Basic Performance

The Fundamental Performance module will familiarize the trainee with plant test procedures, test equipment, and testing as well as plant records, reports, and data collection. It will provide a basic understanding of thermodynamics used in testing plant heat transfer.

(d) Plant Familiarization

The Plant Familiarization module will provide for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the plant.

(4) Health Physics and Chemistry Initial Training

(a) Enrichment Plant Fundamentals

The Enrichment Plant Fundamentals module will provide the new employees with required material on administration, mathematics, physical science, systems, and safety.

(b) Fundamental Health Physics

The Fundamental Health Physics module will present to the students a more comprehensive and theoretical understanding of the nuclear processes with which they are involved. In addition, the techniques for applying theory will be presented in this module. Use will be made of various non-automated counting and spectrographic equipment and portable survey instruments.

(c) Fundamental Chemistry

The Fundamental Chemistry module will provide familiarization with chemistry theory, techniques, and procedures. The overall goal of this module will be familiarization necessary for chemistry technicians to be able to work safely and competently in the enrichment plant.

(d) Plant Familiarization

The Plant Familiarization module will provide for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the plant.

(5) Engineer/Professional Initial Training

This training is part of the Technical Staff and Managers Training Program, and will consist of the following:

(a) Facility Orientation

The Facility Orientation module will provide an orientation to each section within the CEC. An on-the-job task list will provide the trainee with training objectives which must be accomplished while working in the section.

(b) Basic Engineer/Professional Training

The Basic Engineer/Professional Training will provide basic understanding of how uranium is enriched, the systems and components required for producing the final product, and the interrelationship of the various facility organizations in achieving the overall objective.

(c) Enrichment/Chemical Engineer/Professional Training

The Enrichment/Chemical Engineer/Professional Training will provide specific theoretical information related to enrichment plant operations. Topics (e.g. Thermal Science, Nuclear Physics) will address applications in an enrichment plant.

(d) Engineer/Professional Systems Training

The Engineer/Professional Systems Training will provide an overview of plant systems, components and procedures necessary to operate an enrichment plant safely.

10.4.4.2 On-the-Job Training and Qualifications

On-the-job training (OJT) is a systematic method of providing the required job related skills and knowledge for a position. This training will be conducted in the work environment. Applicable tasks and related procedures for each technical area will make up the OJT/qualifications program which will be designed to supplement and complement training received through formal classroom, laboratory, and/or simulator training. The objective of the program is to ensure the trainees ability to perform job tasks as described in the task descriptions and Training Qualifications Guides.

10.4.4.3 Continuing Training

Apart from Initial Qualification and Basic Training, Continuing Training is any training which will maintain and improve job-related knowledge and skills such as the following:

- Facility Systems and Component Changes

- On-the-Job Training/Qualifications Program Retraining

- Procedure and Directive Changes

- Operating Experience Program Documents Review to include Industry and In-House Operating Experiences

- Continuing Training required by Regulation (for example, Emergency Plan Training)

- General Employee, Special, Administrative, Vendor, and/or Advanced Training topics supporting tasks which are elective in nature

- Training identified to resolve deficiencies (task-based) or to reinforce seldom used knowledge skills

- Refresher training on initial training topics

- Pre-job instruction, mock-up training, walk through, that are structured

- Quality Awareness

Each Section's Continuing Training Program will be developed from a systematic approach, using information from job performance and safe operation as a basis for determining the content of continuing training.

Once the objectives for Continuing Training have been established, the methods for conducting the training may vary. The method selected will provide clear evidence of objective accomplishment and consistency in delivery.

10.4.4.4 Special Training

Special training will involve those subjects of a unique nature required for a particular area of work. Special training will usually be given to selected personnel based on specific needs not directly related to disciplinary lines.

10.4.5 Training Program Evaluations

Training and qualifications activities will be monitored by designated facility personnel, the Production Training Services staff, and by Employee Training and Qualifications Services (ETQS) Working Groups. The Quality Assurance Department will audit the facility Employee Training and Qualification System. In addition, trainees and vendors will be encouraged to provide input concerning training program effectiveness. Methods utilized to obtain this information will include, among other things, surveys, questionnaires, performance appraisals, staff evaluation, and overall training program effectiveness evaluation. Frequently conducted classes will not be evaluated each time. However, they will be routinely evaluated at a frequency sufficient to determine program effectiveness. Evaluation information will be collected through:

- verification of program objectives as related to job duties for which intended
- periodic Working Group Program evaluations
- testing to determine student accomplishment of objectives
- student evaluation of the instruction
- supervisor's evaluation of the trainee's performance after training on-the-job
- supervisor's evaluation of the instruction.

Unacceptable individual performance will be transmitted to the appropriate group Superintendent.

10.4.6 Training Records

Accurate records will be maintained on each employee's qualifications, experience, training and retraining. The employee training file shall include records of all general employee training, technical training, and employee development training conducted at the CEC. The employee training file shall also contain records of special company sponsored training conducted by others. The training records for each individual will be maintained so that they are accurate and retrievable. Records of training, qualification, and requalification, as required by Section 2.5 of the applicant's PLCs (LES, 1993e), will be retained for the duration of the facility license.

10.4.7 Conclusion

The NRC staff finds the training program to be adequate, and agrees with the applicant's commitment, by license conditions, to establish planned training programs for CEC employees, addressing all safety disciplines, ALARA practices, issues related to 10 CFR Part 19 (NRC, 1973a), environmental protection, and emergency procedures. The staff agrees with the applicant's PLC that the training program shall also include in depth training to be provided to individuals, depending on job requirements in radiological safety and criticality safety controls.

In addition, the NRC staff agrees with the applicant's commitments concerning training requirements for visitors and contractors, and commitments concerning the retention of training, qualification, and requalification records.

10.5 Operating Procedures

Activities involving licensed materials will be conducted through the use of approved, written procedures. Applicable procedure and training requirements will be satisfied before use of the procedure (LES, 1993e). Before initial enrichment activities occur at the facility, a list of titles of procedures that clearly indicate their purpose and applicability will be made available to the NRC for their inspection. Procedures will be used to control activities in order to ensure the activities are carried out in a safe manner. Major activities to be addressed by procedure (LES, 1993a) include:

- cylinder handling
- autoclave operation
- takeoff stations operation
- other production operations (e.g., blending)
- implementing the Fundamental Nuclear Material Control (FNMC) Plan
- implementing the Emergency Plan
- implementing the Health physics Program
- implementing the Environmental Monitoring Program
- implementing the Physical Security Plan
- implementing the Security Plan for the Protection of Classified Matter and Information
- design changes to the facility
- maintenance of facility structures, systems and components
- construction and testing of facility structures systems and components
- implementing the Quality Assurance Program
- training
- criticality safety.

The NRC staff agrees with applicant's PLC that activities involving licensed materials shall be conducted through the use of approved written procedures, and that applicable procedure and training requirements shall be satisfied prior to receipt of licensed material. Also by license condition: (1) all procedures shall be reviewed for adequacy biennially; (2) all new procedures or changes to existing procedures shall be subjected to the safety evaluation requirements as described in Section 1.5.1 of the applicant's PLCs (LES, 1993e).

10.5.1 Preparation of Procedures

For operating, abnormal, maintenance, instrument, periodic test, chemistry, radioactive waste management, health physics, emergency preparedness, annunciator responses, and modification procedures, each procedure will be assigned to a member of the facility staff for development. Initial procedure drafts will be independently reviewed by members of the facility staff, and/or by personnel from the supplier of centrifuges (Urenco), and/or other vendors. Procedures important to safety will be subjected to an independent review. If a procedure involves QA directly, the QA Manager also must approve the procedure. QA will perform performance based audits to ensure procedure effectiveness.

The NRC staff agrees with applicant's PLCs which state: (1) Procedures shall be prepared , reviewed, and approved in accordance with written procedure requirements; (2) Procedures shall identify limits and controls important to safety and environmental protection; (3) Maintenance and testing, including calibration procedures shall be written and implemented for structures, systems and components important to safety; (4) Procedures important to safety shall be subjected to an independent review. The designated approver shall determine whether or not any additional, cross-disciplinary review is required; (5) Policies shall be developed and implemented for an integrated approach to procedure development and approval; and (6) The CEC Manager or designee shall approve all procedures.

10.5.2 Administrative Procedures

Facility administrative procedures (Department Directives) will be written by each department as necessary to control facility testing, maintenance, and operating activities. Listed below are several areas for which administrative procedures will be written, including principal features:

- Operator's authority and responsibility: The operator will be given the authority to manipulate controls which directly or indirectly affect the enrichment process, including a shut down of the process if deemed necessary by the Shift Supervisor. The operators will also be assigned the responsibility for knowing the limits and set points associated with safety equipment and systems as specified in designated operating procedures

- Activities affecting facility operation or operating indications: All facility personnel performing functions which may affect unit operation or control room indications will be required to notify the Control Room Operator (operator) prior to initiating such action. Removal of an instrument or component from service will require the permission of the Shift Supervisor or Unit Supervisor

- Manipulation of facility control: No one will be permitted to manipulate the facility controls who is not an operator, except for operator trainees under the direction of a qualified operator

- Relief of Duties: This procedure will provide a detailed checklist of applicable items for shift turnover

- Equipment control: Equipment control will be maintained and documented through the use of tags, labels, stamps, status logs or other suitable means

- Master surveillance testing schedule: This procedure will establish a master surveillance testing schedule to ensure that required testing is performed and evaluated on a timely basis. Surveillance testing will be scheduled such that the safety of the facility is not dependent on the performance of a structure, system or component which has not been tested within its specified testing interval. The master surveillance testing schedule will identify surveillance and testing requirements, applicable procedures, and required test frequency. Assignment of responsibility for these requirements will also be indicated

- Control Room Operations Logbook: This logbook will contain significant events during each shift such as enrichment changes, alarms received, or abnormal operational conditions

- Fire Protection Procedures: These procedures will be written to address such topics as training of the fire brigade, reporting of fires, and control of fire stops. The facility's Industrial Safety group will have responsibility for fire protection procedures in general, with the facility's maintenance section having responsibility for certain fire protection procedures such as control of repairs to facility fire stops.

Activities which affect the proper functioning of the facility's Quality Assurance Level 1 or 2 systems and components will be performed in accordance with approved, written procedures. These procedures will be intended to provide a pre-planned method of conducting operations of systems, in order to eliminate errors due to on-the-spot analysis and judgements.

All procedures will be sufficiently detailed that qualified individuals can perform the required functions without direct supervision.

Typical operating activities to be addressed by procedures or checklists are:

- Functional Test of a Cascade
- Evacuation and Preparatory Work Before Run Up of a Cascade

- Run Up of a Cascade
- Run Down of a Cascade
- Calibration of Pressure Transmitter
- Taking UF_6 Samples of a Cascade
- Installation of UF_6 Cylinders in Autoclaves and Preparation for Operation
- Removal of UF_6 Cylinder from Autoclaves
- Installation of UF_6 Cylinders in UF_6 Take Off Stations
- UF_6 Gas Sampling in Take off Lines
- UF_6 Sampling in Homogenizing Autoclaves
- Emptying of Desublimer
- Exchange of Chemical Traps in Vent Systems

Plant specific procedures which address abnormal operating conditions will be written for the CEC. These procedures will be based on a sequence of observations and actions, with emphasis placed on operator responses to indications in the Control Room. When immediate operator actions are required to prevent or mitigate the consequences of an abnormal situation, procedures will require that those actions be implemented at the earliest possible time, even if full knowledge of the abnormal situation is not yet available. The actions outlined in abnormal operating condition procedures will be based on a conservative course of action to be followed by the operating crew.

Typical abnormal operating condition procedures (consisting of appropriate subprocedures) are:

- Power Failure
- Loss of Heat Tracing
- Damaged UF_6 Cylinder Repairs

Temporary operating procedures will be approved written procedures issued for operating activities which are of a nonrecurring nature. Examples of such uses are:

- to direct operating activities during special testing or maintenance

- to provide guidance in unusual situations not within the scope of normal procedures

- to ensure orderly and uniform operations for short periods of time when the facility, a unit, a cascade, a structure, a system or a component is performing in a manner not addressed by existing procedures or has been modified in such a manner that portions of existing procedures do not apply.

The format of these procedures will include a purpose, limits and precautions, initial conditions and step-by-step instructions for each mode of operation and necessary enclosures.

Temporary operating procedures will be sufficiently detailed that qualified individuals can perform the required functions without direct supervision.

Annunciator response procedures will be written which specify operator actions necessary to respond to an off-normal condition as indicated by an alarm. The format for annunciator response procedures will include alarm set points, probable causes, automatic actions, immediate manual actions, supplementary actions, and applicable references.

Maintenance of facility structures, systems and components will be performed in accordance with written procedures, documented instructions, checklists, or drawings which conform to applicable codes, standards, specifications, and criteria. Where appropriate sections of related vendor manuals, instructions or approved drawings with acceptable tolerances do not provide adequate guidance to assure the required quality of work, an approved, written maintenance procedure will be provided.

The facility's maintenance group under the Superintendent of Maintenance will have responsibility for preparation and implementation of maintenance procedures.

Maintenance, testing and calibration of facility instruments important to safety will be performed in accordance with approved written procedures.

Testing conducted on a periodic basis to determine various facility parameters and to verify the continuing capability of structures, systems and components important to safety to meet performance requirements will be conducted in accordance with approved, written procedures. Periodic test procedures will be utilized to perform such testing and will be sufficiently detailed that qualified personnel can perform the required functions without direct supervision.

Periodic test procedures will be performed by the facility's Technical Support, Operations and Maintenance groups.

Chemical and radiochemical activities associated with facility structures, systems and components important to safety will be performed in accordance with approved, written procedures. The facility's chemistry section will have responsibility for preparation and implementation of chemistry procedures.

Radioactive waste management activities associated with the facility's liquid, gaseous, and solid waste systems will be performed in accordance with approved written procedures. The facility's operations, chemistry and health physics sections will have responsibility for preparation and implementation of the radioactive waste management procedures.

10.5.3 Changes to Procedures

Section 2.6.6 of the PLCs (LES, 1993e) and SAR Section 11.4.1.4 (LES, 1993a) assert that changes to facility procedures will be processed by methods described below.

(a) The preparer will document the change as well as the reason for the change.

(b) A safety evaluation will be performed as specified in Section 1.5.1 of the PLCs (LES, 1993e). If the safety evaluation reveals that a change to the license is needed to implement the proposed changes, the change will not be implemented until prior approval from the NRC.

(c) The procedure with proposed changes will be reviewed by a qualified reviewer.

(d) The CEC Manager, a superintendent, or a designee approved by the CEC Manager will be responsible for approving procedure changes, and for determining whether a cross-disciplinary review is necessary, and by which group(s). The independent review will be by an individual outside the group (that is, responsibility of a different manager) that prepared the procedure change. The need for the following cross-disciplinary reviews will be considered, as a minimum:

- For proposed changes having a potential impact on chemical or radiation safety, a review will be performed for chemical and radiation hazards, including radiological effluents. Approved changes will be so indicated in writing by the Technical Support Superintendent or designee

- For proposed changes having a potential impact on criticality safety, a criticality safety review will be performed. Approved changes will be so indicated in writing by the Technical Support Superintendent or designee

- For proposed changes potentially affecting Material Control and Accounting, a material control review will be performed

- The approver of a procedure change will be responsible for ensuring the procedure change was prepared and reviewed by qualified individuals.

(e) Records of completed cross-functional reviews will be maintained in accordance with Section 2.9 of the PLCs (LES, 1993e) for all changes to procedures important to safety.

10.5.4 Distribution of Procedures

Section 2.6.3 of the PLCs (LES, 1993e) and Section 11.4.1.5 of the SAR (LES, 1993a) assert that originally issued, approved procedures and procedure revisions will be distributed in a controlled manner.

The CEC will establish and maintain an index of the distribution of copies of all facility procedures and manuals (for example, the Department Directives Manual, Operations and Maintenance Manuals, etc.). Revisions to facility manuals will be controlled and distributed in accordance with this index. Facility manual indexes will be reviewed and updated on a periodic basis or as required.

Superintendents or their designees shall be responsible for ensuring that all personnel doing work which requires the use of procedures have ready access to controlled copies of the procedures.

10.5.5 Conclusion

The NRC staff finds adequate the applicant's proposals regarding procedure preparation requirements, description, and change and distribution methodologies.

10.6 Records

Section 11.4.2 of the SAR (LES, 1993a) asserts that records management will be in a controlled and systematic manner in order to provide identifiable and retrievable documentation.

The CEC will maintain a Master File which will have controlled access and use. Documents in the Master File will be legible and will be identifiable as to the subject to which they pertain. Documents will be considered valid only if stamped, initialed, signed or otherwise authenticated and dated by authorized personnel. Documents in the Master File may be original copies or reproduced copies. Computer storage of data may be used in the facility Master File.

In order to preclude deterioration of records in the Master File, the following requirements will be applicable:

(a) Records shall not be stored loosely. Records shall be firmly attached in binders or placed in folders or envelopes. Records should be stored in steel file cabinets.

(b) Special processed records, for example, radiographs, photographs, negatives, microfilm, etc., which are light-sensitive, pressure-sensitive and/or temperature-sensitive, shall be packaged and stored as recommended by the manufacturer of these materials.

(c) Computer storage of records shall be done in a manner to preclude inadvertent loss and to ensure accurate and timely retrieval of data.

A Master File storage system will provide for the accurate retrieval of information without undue delay. Written instructions will be prepared regarding the storage of records in a Master File, and a supervisor will be designated with the responsibility for implementing the requirements of the instructions. These instructions will include, but not necessarily be limited to, the following:

(a) A description of the location(s) of the Master File and an identification of the location(s) of the various record types within the Master File

(b) The filing system to be used

(c) A method for verifying that records received are in agreement with any applicable transmittal documents and are in good condition. This is not required for documents generated within a section for use and storage in the same sections' satellite files

(d) A method for maintaining a record of the records received

(e) The criteria governing access to and control of the Master File

(f) A method for maintaining control of and accountability for records removed from the Master File

(g) A method for filing supplemental information and for disposing of superseded records.

A qualified Fire Protection Engineer will evaluate record storage areas (including satellite files) to assure records are adequately protected from damage. The Fire Protection Engineer will be a registered Professional Engineer qualified for membership grade status in the Society of Fire Protection Engineers.

Section 2.9 of the PLCs (LES, 1993e) asserts that records related to health and safety shall be maintained in accordance with the requirements of Title 10 of the Code of Federal Regulations. Records shall be stored to permit easy retrievability. These records document the quality of items and activities performed at the CEC and shall be stored in cabinets or storage facilities which protect the records from damage from fire, water, dust, extreme humidity, and extreme temperatures. Records of instrument calibrations, changes to procedures, audits and inspections, and ALARA findings shall be retained for at least three years. The following records shall be retained for the duration of the license:

- Records of reportable events
- Records and drawing changes reflecting design modifications made to systems and equipment important to safety
- Records of radioactive shipments
- Records of radiation exposure for all individuals entering radiation control areas
- Records of gaseous and liquid radioactive and hazardous material released to the environs
- Records of training, qualification, and requalification as required by Section 2.5 of the PLCs (LES, 1993e) for current and past members of the CEC staff
- Records of safety evaluations described in Section 1.5.1 of the PLCs (LES, 1993e)

- Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analyses at a later date
- Records of QA activities required by the QA Program. These shall be retained for a period of time as recommended by NQA-1-1989
- Records of plant radiation surveys and environmental surveys
- Records of FSRC activities.

The NRC staff agrees with the applicant's records-keeping program described in Section 11.4.2 of the SAR (LES, 1993a) and with the commitments proposed in Section 2.9 of the applicant's PLCs (LES, 1993e).

10.7 Management Controls

Section 11.4.3 of the SAR (LES, 1993a) asserts that a review and audit program for operational quality assurance of the CEC will be established, and periodically reviewed by management, to:

- verify that facility operations are consistent with LES company policy, approved procedures and license provisions

- review important proposed facility modifications, tests and procedures

- verify that reportable occurrences are investigated and corrected in a manner which reduces the probability of recurrence of such events (reference SER Section 10.7.6)

- to detect trends which may not be apparent to a day-to-day observer.

The intent of this program will be to ascertain that the facility is constructed and operated safely and in accordance with the license conditions. The organizational structure for conducting the operational quality assurance review and audit program will be the FSRC and regular audits conducted by the Quality Assurance Department.

10.7.1 Facility Safety Review Committee

Section 2.3 of the PLCs (LES, 1993e) and Section 11.4.3.1 of the SAR (LES, 1993a) assert that the FSRC shall report to the CEC Manager, and shall provide technical and administrative review of CEC operations which could impact plant worker and public safety. The scope of activities reviewed by the FSRC shall, as a minimum, include the following safety activities and practices:

- Quality Assurance
- Radiological protection
- Nuclear criticality safety

- Chemical safety
- Industrial safety including fire protection
- Environmental protection
- ALARA policy implementation
- Changes in facility design or operations
- Training Programs
- Incident reports, including root cause evaluations, and violations of regulations or license conditions.

The FSRC will conduct at least one facility audit (that is, review) per year for the above areas. This audit/review will be a management assessment type review, not an audit that is performed by QA personnel.

The FSRC shall be composed of at least 5 members, including the Director. Members of the FSRC may be from the LES corporate office or CEC technical staff. The 5 members shall include experts on operations and all safety disciplines (criticality, radiological, chemical, industrial). The Director, members and alternate members of the FSRC shall be formally appointed by the CEC Manager; shall have an academic degree in an engineering or physical science field; and, in addition, shall have a minimum of 3 years of technical experience, of which a minimum of 1 year shall relate directly to one or more of the safety disciplines (criticality, radiological, chemical, industrial). At least one member of the FSRC shall have the qualifications of the Projects Analyst described under the Project Manager subtitle in SER Section 10.2. Members of the FSRC shall receive training on possible error modes of management systems.

The FSRC shall meet at least once per calendar quarter during the period of initial operation. Subsequently, the meeting frequency shall not be less than three (3) each calendar year with a maximum interval of 180 days between any two consecutive meetings.

Review meetings shall be held within 60 days of any incident which is reportable to the NRC. These meetings may be combined with regular meetings. Following a reportable incident, the FSRC shall review the incident's causes, the responses, and both specific and generic corrective actions to ensure resolution of the problem is implemented.

A written report of each FSRC meeting and review shall be forwarded to the station manager and superintendents within 30 days and be retained for the duration of the facility license.

As stated in SER Section 10.6, records of FSRC activities shall be maintained for the life of the facility.

10.7.2 Quality Assurance Department

The Quality Assurance Department shall conduct periodic audits and inspections and other activities associated with the CEC, in order to verify the facility's compliance with established requirements as detailed in SER Chapter 12.

10.7.3 CEC Operating Organization

Section 11.4.3.3 of the SAR (LES, 1993a) asserts that the facility operating organization will provide, as part of the normal duties of supervisory personnel, timely and continuing monitoring of operating activities to assist the CEC Manager in keeping abreast of general facility conditions and to verify that the day-to-day operating activities are conducted safely and in accordance with applicable administrative controls.

These continuing monitoring activities are an integral part of the routine supervisory function and are important to the safety of the facility operation. The CEC operating organization, staff qualifications, functions, and responsibilities are detailed in SER Sections 10.2 and 10.3.

10.7.4 CEC Audited Organizations

Section 11.4.3.4 of the SAR (LES, 1993a) asserts that audited organizations shall assure that deficiencies identified are corrected in a timely manner.

Audited organizations will transmit a response to each audit report within the time period specified in the audit. For each identified deficiency, the response will identify the corrective action taken or to be taken. For each identified deficiency, the response will also address whether or not the deficiency is considered to be indicative of other problems (for example, a specific audit finding may indicate a generic problem). With regard to corrective action to be completed at some future date, the audited organization will notify the auditing organization of the date of completion of the committed corrective action within 30 days thereof. Other supplementary response information may be provided as appropriate.

10.7.5 Internal Audits and Inspections

Audits and inspections shall be performed to assure that plant operations are conducted in accordance with the operating procedures.

Section 2.7 of the PLCs (LES, 1993e) asserts that audits and inspections shall be conducted by Quality Assurance group personnel and other individuals technically qualified to perform audits and inspections to determine that plant operations are conducted in compliance with regulatory requirements, license conditions, and written procedures. These audits and inspections shall be the responsibility of the QA Manager. As a minimum, they shall assess programs and activities related to:

- preventive maintenance
- training
- emergency planning
- radiation protection
- criticality safety control
- hazardous chemical safety
- fire protection
- environmental protection
- quality assurance activities performed by personnel outside the QA organization.

Audits shall be performed in accordance with a written plan which will identify and schedule audits to be performed. Audit team members shall not have direct responsibility for the function and area being audited, shall have technical expertise and experience in the area being audited, and shall be indoctrinated in audit techniques. Audits shall be conducted on an annual basis.

The results of the audits shall be provided in a written report within 30 days of the audit to the CEC Manager, the FSRC, and the superintendent responsible or the activities audited. Any deficiencies noted in the audit shall be responded to in writing by the superintendent or designee within 30 days, tracked to completion by an individual designated by the QA organization, and re-examined during future audits to ensure corrective action has been completed.

Inspections shall be performed routinely by qualified staff personnel that are not directly responsible for production activities being inspected. Inspections shall be conducted at least semi-annually. Deficiencies noted during the inspection requiring corrective action shall be forwarded to the superintendent or designee of the applicable area or function for action. The responsible superintendent, or designee, shall respond in writing to deficiencies noted in inspections. Future inspections shall include a review to evaluate if corrective actions have been effective.

Inspections shall be performed in accordance with a written plan by qualified staff personnel that are not directly responsible for activities being inspected. Inspections shall be conducted in accordance with a written plan with a frequency commensurate with the activity being inspected. Deficiencies noted during the inspection requiring corrective action shall be documented in a written report going to the appropriate level of management for follow-up action. Future inspections shall include a review to evaluate if corrective actions have been effective.

The NRC staff agrees with the applicant's internal audits and inspections programs, and with the commitments stated in Section 2.7 of the PLCs (LES, 1993e).

10.7.6 Investigations and Reporting

Section 11.4.5 of the SAR (LES, 1993a) asserts that unusual events which potentially threaten or lessen the effectiveness of health, safety or environmental protection will be identified and reported to and investigated by the Compliance Superintendent. Each event will be considered in terms of its requirements for reporting in accordance with regulations and will be evaluated to determine the level of investigation required.

These evaluations and investigations will be conducted in accordance with approved procedures. The depth of the investigation will depend upon the severity of the classified incident in terms of the levels of uranium released and/or the degree of potential for exposure of workers, the public or the environment.

The Compliance Superintendent is responsible for (1) maintaining a list of agencies to be notified, (2) determining if a report to an agency is required, and (3) notifying the agency when required. The licensing function has the responsibility for continuing communications with government agencies and tracking corrective actions to completion.

Section 2.8 of the PLCs (LES, 1993e) asserts that the process of incident identification, investigation, root cause analysis, environmental protection analysis, recording, reporting, and follow-up shall be addressed in and performed by written procedures. Radiological, criticality, hazardous chemical, and industrial safety requirements shall be addressed. Guidance for classifying occurrences shall be contained in facility procedures, including a list of threshold off-normal occurrences.

The Compliance Superintendent shall maintain a record of corrective actions to be implemented as a result of off-normal occurrence investigations. These corrective actions shall include documenting lessons learned, and implementing worker training where indicated, and shall be tracked to completion by the Compliance Superintendent or designee.

The NRC staff agrees with the applicant's investigation and reporting program.

10.7.7 Modifications to Facilities and Equipment

The applicant asserts in Section 11.4.6 of the SAR (LES, 1993a) that in order to provide for the continued safe operation of the CEC structures, systems and components, measures will be implemented to ensure that the quality of these structures, systems and components is not compromised by planned changes (modifications). After issuance of the Facility Construction/Operating License, the CEC Manager will be responsible for the design of and modifications to facility structures, systems or components. The design and implementation of modifications will be performed in a manner so as to assure quality is maintained in a manner commensurate with the remainder of the system which is being modified, or as dictated by applicable regulations.

The administrative instructions for modification will be contained in the "CEC Facility Modification Manual" which will be approved, including revisions, by the CEC Manager with concurrence of the Manager of Quality Assurance. The manual will contain the following items necessary to ensure quality in the modification program:

- The requirements which shall be met to implement a modification
- The requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The CEC Facility Modification Manual shall be written to ensure that policies are formulated and maintained to satisfy the quality assurance standards specified in 10 CFR Part 50, Appendix B, as applicable.

Each change to the facility shall have a safety evaluation performed in accordance with the CEC License. Each modification will also be evaluated for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents.

Each modification will also be evaluated and documented for radiation exposure, to minimize worker exposures in keeping with the facility ALARA program, criticality and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include, but will not be limited to the review of: modification cost, similar completed modifications, QA aspects, potential operability or maintainability concerns, constructability concerns, post-modification testing requirements, environmental considerations, and human factors.

After completion of a modification to a structure, system, or component, the Projects Manager, or designee, will ensure that all appropriate testing has been completed to ensure correct operation of the system(s) affected by the modification and documentation regarding the modification is complete. In order to ensure operators are able to operate a modified system safely, when a modification is complete, all documents necessary, i.e., the revised process description, checklists for operation and flowsheets will be made available to operations and maintenance before the modified system becomes "operational." A formal notice of a modification being completed will be distributed to all Superintendents within 5 working days. For modifications to Quality Assurance Level 1 or 2 systems, structures, or components, as-built drawings incorporating the modification will be completed within six months.

The NRC staff is in agreement with the applicant's equipment and facility modification program.

10.8 Emergency Planning

Section 70.22(i)(1) of Title 10 of the Code of Federal Regulations requires that each application to possess enriched uranium or uranium hexafluoride in the amounts LES proposes must contain either,

(a) an evaluation showing that the maximum dose to a member of the public offsite as a result of release of radioactive materials would not (1) exceed one rem effective dose equivalent, or (2) involve an intake of more than two milligrams of soluble uranium, or

(b) an emergency plan for responding to the radiological hazards of an accidental release of special nuclear material and to any chemical hazards directly incident thereto.

LES elected to include an emergency plan in its application and the NRC staff has reviewed the applicant's Emergency Plan using the guidance of Regulatory Guide 3.67, Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities. The emergency plan content requirements in 10 CFR 40.31(j)(3) and 10 CFR 70.22(i)(3) are identical and NRC staff's evaluation of those requirements is as follows:

(i) Facility Description. Chapter 1 of the Emergency Plan adequately describes the licensed activity, the site, and the area near the site.

(ii) Types of Accidents. Chapter 2 of the Emergency Plan adequately identifies each type of radioactive materials accident for which protective actions may be needed. The maximum exposures to offsite individuals are based on the analysis in Chapter 9 of the applicant's SAR which is evaluated in Chapter 11 of this SER.

(iii) Classification of Accidents. Chapter 3 of the Emergency Plan establishes an adequate system for classifying accidents as an Alert or Site Area Emergency. The Emergency Plan requires a Site Area Emergency to be declared for a release involving more than 100 kilograms of UF_6. An Alert will be declared for a release involving more than 1 kilogram, but less than 100 kilograms of UF_6. Specific emergency action levels for declaring an Alert or Site Area Emergency will be established in the implementing procedures.

(iv) Detection of Accidents. Chapter 2 of the Emergency Plan adequately identifies the means of detecting each type of accident in a timely manner. A UF_6 release will be detected by pressure/temperature monitors or direct observation by workers. A criticality monitor system is provided to detect the initial burst of radiation from a criticality and a fire detection system is also provided.

(v) Mitigation of Consequences. Chapters 5 and 6 of the Emergency Plan contain an adequate description of the means and equipment for mitigating the consequences of each type of accident. Systems are provided for stopping operations, evacuating UF_6 process piping, and extinguishing fires. Procedures are provided for evacuating the plant, controlling contamination, treating injured workers, and recommended protective actions for offsite areas.

Section 7.6 of the Emergency Plan provides an adequate description of the program for maintaining the emergency response equipment.

(vi) Assessment of Releases. Sections 5.2 and 6.4 of the Emergency Plan provide an adequate description of the methods and equipment for assessing releases of radioactive material. The plan provides for water, air, and soil sampling to assess releases and bioassay sampling to assess personnel exposures.

(vii) Responsibilities. Chapter 4 of the Emergency Plan provides an adequate description of the responsibilities of licensee personnel during an emergency. During any emergency, the Shift Supervisor in the Central Control Room acts as the emergency coordinator responsible for directing the response effort until the Plant Manager arrives and takes over. Personnel responsible for maintaining and updating the Emergency Plan are identified in Chapter 7 of the Emergency Plan.

(viii) Notification and Coordination. Chapter 3 of the Emergency Plan provides a clear commitment to promptly notify offsite response organizations of an emergency including notification of the NRC Operations Center immediately after calling the offsite response organizations, but no later than one hour after declaring an emergency. Sections 4.3 and 4.4 provide an adequate description of provisions for assistance from offsite response organizations. Sections 5.6 and 5.7 provide an adequate description of provisions for medical treatment of contaminated workers.

(ix) Information to be Communicated. Section 3.3 of the Emergency Plan provides an adequate description of the type of information to be given to offsite response organizations during an emergency.

(x) Training. Sections 7.2 and 7.3 of the Emergency Plan provide an adequate description of the training the licensee will provide to workers on how to respond to an emergency. All workers receive general safety training and emergency response personnel receive additional training annually. Facility tours and classroom training is also provided to offsite response organizations.

(xi) Safe Shutdown. Chapter 9 of the Emergency Plan provides an adequate description of the means of restoring the facility to a safe condition after an accident.

(xii) Exercises. Sections 7.3 and 7.4 of the Emergency Plan provide adequate provisions for biennial exercises. Offsite organizations are invited to participate and each exercise is critiqued. The Emergency Preparedness Manager is responsible for tracking deficiencies and ensuring that corrective actions are implemented. Section 7.8 of the Emergency Plan includes an adequate provision for quarterly communications checks.

(xiii) Hazardous Chemicals. Chapter 10 of the Emergency Plan provides an adequate certification that the applicant will meet its responsibilities under the Emergency Planning and Community Right-To-Know Act of 1986.

In addition to the plan contents, the introduction to the Emergency Plan identifies the offsite organizations that were allowed to review the Emergency Plan pursuant to the requirement in 10 CFR 40.31(j)(4) and 10 CFR 70.22(i)(4). Agreement letters with the offsite organizations are provided in Chapter 11 of the Emergency Plan.

The NRC staff finds that the Emergency Plan demonstrates that an acceptable program has been established for responding to the radiological hazards of an accident involving licensed material and to any associated chemical hazards directly incident thereto.

10.9 Start-up and Inspections

The NRC staff will conduct an inspection of the CEC prior to start-up to verify that the initial construction is in accordance with the requirements of the license. The NRC staff will monitor construction, preoperational tests, start-up tests and operations at the CEC in accordance with NRC regulations.

10.10 Conclusion

The NRC staff concludes that the programs proposed by the applicant, Louisiana Energy Services, are adequate and meet the requirements specified in 10 CFR Parts 40 and 70 concerning: (1) information on the control or ownership of the applicant; (2) organizational structure for the CEC; (3) CEC staff technical qualifications, functions, and responsibilities; (4) training, (5) operating procedures; (6) records management; and (7) management controls. The NRC staff also finds acceptable the CEC Emergency Plan, submitted as part of the application.

11 ACCIDENT ANALYSIS

Operation of the Claiborne Enrichment Center (CEC) introduces additional risk to worker and public health and safety because of possible accidents and their potential consequences: personal injury, health effects from acute exposure to toxic chemicals, non-stochastic effects from acute radiation exposure, and risk of latent cancer because of exposure to radioactive material. The purpose of this analysis is to investigate the nature and consequences of possible CEC accidents, to support a finding on the impact of the facility on public health and safety, and to review and identify limiting conditions for operation. The first section summarizes the methods used in the analysis and past NRC research on the toxicity of uranium hexafluoride (UF_6). The second section discusses potential hazards related to the CEC and summarizes the results of a hazard audit. The third section describes potential CEC accidents and evaluates the consequences of occurrence of these accidents. The fourth section summarizes the results of the accident analysis. The analysis presented in this chapter is independent NRC staff analysis.

11.1 Methods of Accident Evaluation

The analytical procedures used in this safety evaluation included identifying and assessing hazards, reviewing potential accident initiators and related release mechanisms, developing accident scenarios, and examining the consequences of occurrence of the selected set of potential accident scenarios. In this context, the term "hazard" means any radiological or chemical substances; any concentration of chemical, electrical, or mechanical energy; or any equipment design or configuration, which could, by itself or in combination, contribute to adverse environmental or worker and public health and safety impact. The basis of the analysis underlying this evaluation is the proposed design of the facility as described in the CEC Environmental Report (ER) (LES, 1993b), Safety Analysis Report (SAR) (LES, 1993a), and applicant responses to NRC Requests for Additional Information (RAIs) (LES, 1993d, 1992a, 1992b, and 1992c), and was of a deterministic, non-probabilistic nature. The hazard audit surveyed all materials used in the facility, inventoried quantities and flows, and considered chemical, physical, and toxicological properties of the materials. Accident initiators and scenarios were identified by reviewing Urenco experience in European centrifuge enrichment plants, past NRC-sponsored evaluations of accident scenarios involving UF_6 at NRC-licensed production and fuel fabrication facilities (Siman-Tov, 1984), and the description of equipment and operations presented in the CEC SAR. Atmospheric dispersion analysis required for evaluation of releases of material was performed in a manner consistent with NRC Regulatory Guide (RG) 1.145, Atmospheric Dispersion Models For Potential Accident Consequences At Nuclear Power Plants (NRC, 1982b). Detailed description of dispersion modeling is presented in Chapter 4. The dispersion modeling established that, for elevated releases, the point of maximum contaminant concentration is located 400 meters north of the plant stacks. This location is outside of the restricted area (fence line) but within the site boundary. In order to provide a conservative analysis, this location is used as the point of maximum exposure for offsite individuals. The potential stochastic radiological consequences of exposure to uranium were quantified as effective dose equivalent by using the methods of ICRP-26 (ICRP, 1977) and ICRP-30 (ICRP, 1980).

The acute radiological and toxic chemical effects of exposure to uranium (U) and hydrogen fluoride (HF) have been previously evaluated by the NRC in NUREG-1391 (NRC, 1991a). Hydrogen fluoride and uranyl fluoride are produced when UF_6 reacts with water. The NRC analysis concluded that the chemical effects of exposure to uranium exceeded the acute radiological effects and that the threshold for clinically observable non-stochastic effects corresponded to an intake of 10 milligrams of uranium. Intakes below this level produce no harmful effects. Therefore, the primary concern for exposure to uranium is from the chemical rather than radiological effects. Similarly, exposure to HF at a concentration of 25 mg/m^3 for 30 minutes was identified as the level for no significant effects, either short-term or long-term. The threshold concentration level for exposure to HF was found to be inversely proportional to the square root of exposure time. By Commission Order (NRC, 1991b), the criteria specified in NUREG-1391 are given the force of a standard to be used for the purposes of siting and design of the CEC against accidental releases of UF_6.

Analysis and calculation methods used in the accident analysis incorporated conservative elements in order to provide a reasonable upper bound to potential impacts. Equipment inventories or process flow rates used in the analysis were the maximum quantities present at the CEC consistent with environmental and process conditions. In release of material to the plant stack, the effect of dilution in building air was not credited. Plate-out of particulate uranium within the Separations Building or deposition in atmospheric transport was not credited, maximizing the portion of released inventory which can reach a receptor location. Activation of fire suppression systems was not credited with terminating scenarios involving fire. The diluting effect of atmospheric dispersion was evaluated on the basis of guidance of RG 1.145 (NRC, 1982b) which directs the selection of conditions which occur less than 5 percent of the time. The combined effect of incorporating conservative elements into the analysis is to bracket variation in conditions and bound estimated impacts of potential accidents.

11.2 Hazard Audit Results

A hazard audit is a structured inventory procedure which identifies materials, equipment, and energy sources which could pose a threat to worker or public health and safety. The hazard audit was completed as a three-step compilation and evaluation process. The first step developed a list of all chemicals, equipment, and concentrations of energy present at the CEC. The second step cross-referenced the list of potential hazards against a list of physical, chemical, and toxicological properties (Sax, 1986) in order to identify those materials with a potential for significant health effects if released in an uncontrolled manner. The list of hazards included potential interactions between chemicals on the basis of chemical reaction and combustion effects. Products of interaction were evaluated against chemical, physical, and toxicological properties on an equivalent basis with chemicals originally present on the list of potential hazards. The third step estimated consequences of a bounding, complete-release event. Materials which posed an insignificant impact at the controlled area fence were eliminated from further consideration.

The hazard audit covered all areas of the CEC, and initial results are summarized by area. Within CEC buildings only the Separations Building has quantities or concentrations of hazards which could pose a potential threat to public health and safety. For areas outside CEC buildings, only the UF_6 storage areas and the diesel fuel storage area present potential threats. The initial lists of potential chemical and equipment and energy hazards for the Separations Building are presented in Tables 11.1 (LES, 1992c) and 11.2, respectively.

The list indicates that the primary chemical hazard present at the CEC is UF_6 and that large quantities of UF_6 are not collocated with significant quantities of other potentially hazardous materials. The largest concentration of potentially hazardous materials other than UF_6 is located in the Chemical Storage Area. Here the risk of fire or explosion with related release of combustion products is mitigated by the presence of relatively small amounts of combustible material. The total heat of combustion of the material stored in this room is on the order of tens of billions of joules, a quantity which, if released, could cause local damage in this room, with minor impact on the rest of the building. Release of potential combustion byproducts, in particular, chlorine, hydrochloric acid, and HF, is considered possible in a fire, and a gross screening analysis warrants further consideration of these materials. Similar considerations apply in reviewing the Mechanical Workshop. Combustion of the material in this area would cause local damage and release a quantity of Freon which could be locally harmful. The impacts of fire in the Mechanical Workshop were not considered further because the potential impacts of this accident would be bounded by occurrence of the similar event in the Chemical Storage Area. Examining the remaining areas of the Separations Building did not identify either hazardous chemicals other than UF_6 present in potentially dangerous quantities or collocation of pairs of potentially dangerous chemicals. Aqueous solutions contaminated with uranium are present in the Liquid Waste Disposal System (LWDS). Spills or overflows from this system would be contained in the Separations Building and are not a significant source of hazard to workers or the public.

The major pieces of equipment identified as potentially hazardous because of the presence of UF_6 and thermal energy are the feed, blending, and sampling autoclaves, as shown in Table 11.2. Electrical equipment, although a potential fire hazard, was not considered likely to cause hazardous releases because of location and fire protection. The energy potential of desublimers was estimated to be of little significance. Mechanical equipment, including transporters and cranes, were considered significant because of their size and weight, and the chemical energy in their fuel.

In summary, the hazard audit identified UF_6 stored and used throughout the plant site, and selected chemical storage areas and mechanical equipment of the Separations Building as potential hazards to be evaluated in more detail in the accident analysis.

Table 11.1 Potential chemical hazards present in the Separations Building

Contaminated Pump Workshop		
Fomblin Oil	38 liters	Toxic
Oily Rags (Fomblin Oil)	0.1 m^3	Flammable/Toxic
Trash	0.1 m^3	Flammable
Decontamination/Oil Recovery Room		
Fomblin Oil	415 liters	Toxic
Freon TF	415 liters	Toxic
Oily rags (Fomblin Oil)	0.1 m^3	Flammable/Toxic
Uranium/Carbonate Sludge	210 liters	Toxic
Activated Carbon	0.1 m^3	Flammable
Anhydrous Sodium Carbonate	0.1 m^3	Hazardous
Clean Pump Workshop		
Fomblin Oil	415 liters	Toxic
Fomblin Grease	10 kg	Toxic
Freon TF	210 liters	Toxic
Oily Rags (Fomblin Oil)	0.1 m^3	Flammable/Toxic
I & E Workshop		
Freon TF	210 liters	Toxic
Petroleum Oil	10 liters	Flammable
Oily Rags (Petroleum Oil)	0.1 m^3	Flammable
General Storage		
Paper Products	140 kg	Flammable
Cloth Products	140 kg	Flammable
Plastic Products	140 kg	Flammable
Chemical Laboratory		
Uranium Hexafluoride (in 1/2 lb. cylinders)	230 kg	Toxic
Carbon Tetrachloride	4 liters	Toxic
Various Chemicals	40 liters	Hazardous/Toxic
Waste Storage		
Trash	450 kg	Flammable
Oily Rags (Fomblin Oil)	0.6 m^3	Flammable/Toxic
Oily Rags (Petroleum Oil)	0.6 m^3	Flammable
Uranium/Carbonate Sludge	415 liters	Toxic
Activated Carbon	1.0 m^3	Flammable
Freon TF	210 liters	Toxic

Table 11.1 Potential chemical hazards present in the Separations Building (continued)

Truck Bay		
Oil (in crane gearbox)	8 liters	Flammable
Diesel Fuel (Truck Tank, on occasion)	400 liters	Flammable
Tank Room		
Oil (in pump housings)	8 liters	Flammable
Laundry		
Bleach	45 kilograms	Hazardous
Detergents	90 kilograms	--
Change Room		
Paper Products	45 kg	Flammable
Cloth Products	230 kg	Flammable
Mechanical Workshop		
Petroleum Oil	210 liters	Flammable
Freon TF	420 liters	Toxic
Acetylene (gas)	3.0 m^3	Flammable
Electric Welding Equipment		Ignition Source
Chemical Storage		
Activated Carbon	450 kg	Flammable
Carbon Tetrachloride	20 liters	Toxic
Freon TF	830 liters	Toxic
Fomblin Oil	415 liters	Toxic
Bleach	210 liter	Hazardous
Detergents	90 kg	Hazardous
Petroleum Oil	230 liters	Flammable
Acetylene (gas)	3 m^3	Flammable
Laboratory Chemicals	380 liters	Hazardous/Toxic
Sodium Chloride	450 kg	--
Forklift	--	--
Environmental Laboratory		
Various Chemicals	80 liters	Hazardous/Toxic
Auxiliary Area		
Freon R-11	3,600 kg	Toxic

Table 11.1 Potential chemical hazards present in the Separations Building (continued)

UF_6 Handling Area		
Uranium Hexafluoride (liquid)	40 tonnes	Toxic
Uranium Hexafluoride (solid)	75 tonnes	Toxic
Blending Area		
Uranium Hexafluoride (liquid)	5 tonnes	Toxic
Uranium Hexafluoride (solid)	5 tonnes	Toxic
Cylinder Handling Area		
Uranium Hexafluoride (solid)	25 tonnes	Toxic

Table 11.2 Potentially hazardous equipment and energy sources

Equipment Name	Operation	Hazard
Autoclaves	Liquify UF_6	Thermal energy
Desublimers	Sublime/desublime UF_6	Thermal energy
Cylinder crane	Lift UF_6 cylinders	Gravitational energy
Rail transporter	Move UF_6 cylinders	Mechanical energy
Straddle carriers	Move UF_6 cylinders	Mechanical energy
Rectifiers	Supply electrical power	Thermal energy

11.3 Accident Analysis

Accident analysis provides a basis for assessing potential threat to public health and safety by identifying and evaluating a set of hypothetical scenarios which span the range of possible CEC accidents. Identification of these accident scenarios was based on review of the CEC design and operating plans and past experience in handling UF_6. Review of multiple data sources gives added assurance that all significant potential accident scenarios were identified. A set of representative scenarios is evaluated in detail. The reviews and evaluations discussed below are in part dependent on analysis presented earlier in this SER. Of particular relevance are the SER Chapter 4 analysis of structures, systems, and components important to safety, nuclear criticality safety, and response to design basis events, and the SER Chapter 5 evaluation of instrumentation and controls.

11.3.1 Potential Accident Initiators and Scenarios

The development of scenarios based on review of the CEC design included evaluation of equipment and procedures against potential failure modes and identified design and operating changes which could prevent the accidents and response actions which could mitigate the consequences. This CEC-specific review was supplemented by a review of Urenco operating experience and of past experience at UF_6 separation, production, and fuel fabrication facilities. The review of reports of events at these facilities emphasized the type of observed failure, the type of initiator, and magnitude of release.

11.3.1.1 Accident Scenarios Developed from Review of the CEC Design

The description of process design, equipment, instruments and controls, and operating procedures presented in the CEC SAR provides a basis for identifying potential accident scenarios. The scenario development procedure involved analyzing the plant by segments defined by process function and performing a simplified failure modes analysis for each segment. Within each plant segment, a list of equipment and instruments and controls was developed, operating procedures were reviewed, and a list of failure modes was developed. This procedure considered system response to generic initiators, such as earthquake, loss of power, and operator error or inaction. The consequences of each potential failure mode were noted. The selected plant segments are Feed and Purification, Separation, Tails and Product Removal, Product Sampling, and Product Blending. For each of the plant segments, SER Chapter 3 describes process equipment, and SER Chapter 5 presents simplified diagrams.

The feed and purification segment of the plant includes the feed autoclaves, purification desublimers and cubicles, and associated valves, piping, and controls. The primary controls are the heater protection circuits of the autoclaves and the state switches for the autoclave, desublimer, and purification cubicle. A state switch is a multifunctional selector which activates control circuits for process elements including valves and pumps. A fundamental failure mode for equipment, valves, or piping is mechanical loss of confinement. Causes of such failures include fatigue, corrosion, mechanical impact, or earthquake acceleration. Mechanical damage or thermal over-pressurization and simultaneous rupture of the feed cylinder and autoclave would produce the largest potential release to the atmosphere. Redundant autoclave temperature and pressure heater controls and equipment construction in accord with the Advance Notice of Proposed Rulemaking (ANPR) (NRC, 1988a) design basis make these events unlikely. Because this event has not occurred and diverse, redundant protection systems are present, the event was not considered credible and is not analyzed in the accident analysis. Failure of connector piping or valves within the autoclave results in release to the Gaseous Effluent Vent System (GEVS), with subsequent minor release to the atmosphere. Temperature and pressure monitors within the autoclave and HF monitors in the GEVS line signal this release. Failure of equipment, valves, or piping external to the autoclave results in release to the atmosphere. The nature of such potential events varies from slow fume releases through worn valve seatings to large releases through ruptured piping. As discussed in SER Section 4.2.3.1, the maximum flow from any single ruptured pipe would be less than 50 gm/s. In either case, the release would be indicated by loss of vacuum in the line, with automatic isolation of the line by valves in the autoclave and in the plant unit header. Isolation of the lines would terminate the release. Operator errors--for example, incorrect selection of state switch position--or inadequate degassing of lines could lead to relatively small releases. Operator-induced over-heating of cylinders or desublimers is prevented by independent protection circuits.

The separation cascade segments of the plant include isolation valves, blocks of centrifuges, and associated controls. The inventory of a single centrifuge is 0.01 kilograms, with a total inventory of 420 kilograms for all cascades. The containment failure of a single centrifuge or even an entire block of centrifuges would produce a release with minor health and safety consequences. Leaks in the cascade equipment are indicated in centrifuge control circuits monitored in the central control room. The lack of large inventory, direct human intervention, and motive force for release make significant releases from the separation section of the plant unlikely.

The tails and product removal segments of the plant include vacuum pumps, cylinder stations, desublimers, and associated controls. Like the feed segment of the plant, loss of confinement in pumps, valves, or piping because of wear or mechanical forces could result in release to the building airspace. Such failures would lead to loss of vacuum in the lines, with air inleakage and automatic isolation of the leak. Over-pressurization of lines because of sensor or pump failure could lead to UF_6 desublimation, line blockage, and rupture. Release would be to the Separations Building air space, and the over-pressurization would initiate isolation of the line. Loss of confinement in the cylinder stations would also result in release to the building airspace but at low rates because UF_6 is primarily in the solid state at these locations. The loss of confinement could include damage to cylinders in an earthquake or small-scale leakage in connector piping. Loss of cooling air to the product stations or cooling water to the tails stations would lead to increase in pressure of the cylinder and line, with automatic shut-down and no release to the atmosphere. Abrupt release of large quantities of UF_6 in the product or tails take-off sections is prevented by the absence of significant driving forces. Pressure in the pipes and vessels is sub-atmospheric, and no significant sources of heat are present. Mistaken operator selection of pump state switches for valve position could lead to increase in pressure in the line, with automatic response by pressure sensors and no UF_6 release. Mistaken operator selection of the product cylinder state switch could lead to increase in pressure in the cylinder inlet line, with pressure sensor response to stop flow. Mistaken operator selection of valve position on the vent desublimer could lead to increase in pressure in the inlet line, with automatic response by pressure sensors to stop flow and preclude release of UF_6.

The product sampling and blending systems use heated autoclaves to liquefy enriched UF_6. Failure modes for these autoclaves are similar to those hypothesized for the feed autoclaves.

Catastrophic failure of a heated cylinder and autoclave would release a large amount of UF_6 to the Separations Building airspace and then to the atmosphere. Line or valve leaks outside the autoclave would also be to the building airspace. For the reasons cited above for the feed autoclave, failure of the blending and sampling autoclaves with simultaneous cylinder failure is not evaluated in the accident analysis. Desublimer failure modes also duplicate those developed for the feed purification system. Inadvertent venting and pump seal leaks are possible failure modes, with release to the atmosphere. Incorrect selection of valve positions with the autoclave, desublimer, or receiver cylinder state switches generally would lead to increase in pipeline pressure, with automatic control system response and no release of UF_6.

Electric power is delivered to the CEC on two independent, redundant overhead 115 kilovolt lines from the Louisiana Power and Light (LP&L) grid system. A standby generator system comprised of two redundant package systems provides power to essential loads in the event of loss of offsite power. In addition, an Uninterruptable Power System (UPS) is provided to allow shutdown in the event of total loss of power. Because of the multiple, redundant design and the design feature of electric heating and fail-closed valves in effluent lines, loss of power at the CEC would not lead to a release of UF_6.

11.3.1.2 Experience at Urenco Facilities

Urenco operates UF_6 centrifuge enrichment plants at Capenhurst, United Kingdom; Almelo, Netherlands; and Gronau, Germany. Because the capacity of these plants is similar to that of the proposed CEC, and the kinds of equipment and design features of these plants are proposed for use at the CEC, the operating experience of these plants is relevant to the CEC safety analysis. Urenco centrifuge enrichment facilities at the three sites have a combined operating history of approximately 32 years. The CEC ER (LES, 1993b), SAR (LES, 1993a), and RAIs (LES, 1992b) describe the off-normal events which have occurred in operating the European plants. This information, which is representative of the type and distribution of off-normal events but is not a complete audit of Urenco operating experience, is summarized in Table 11.3.

The data indicate that leaks from disconnecting piping and from pump failures are the most frequent events which lead to UF_6 releases, with estimated frequencies of two times per year and estimated releases on the order of tens of grams. Line disconnection losses have occurred with mobile pump set equipment, with pump maintenance, and with sampling manifold handling. Operator error related to degassing lines and handling sampling manifolds plays a role in this type of event. Pump failures resulted from failure of seals on rotating equipment, in some instances, related to blockage of the pump exhaust line. Off-normal events in the autoclaves have been limited to small leaks in the flexible piping and in valves in the cylinder exhaust line. Response to the in-plant leaks has included revision of operating procedures and training of workers. Off-normal events related to cylinders were limited to damage to cylinder valves from collision with handling equipment and leaks from cylinder valve packing nuts. Resulting leaks were temporarily sealed with tape before replacement of the damaged valve. Response to the events included reconfiguration of the cylinder storage area and redesign of the packing nuts.

Table 11.3 Urenco operating experience with UF_6 leaks

Type of Incident	Number of Incidents	Cause	Response
Line connect/ disconnect leak	5	Inadequate degassing of lines	Revise operating and maintenance procedures
Pump seal failure	5	Mechanical failure of seals	Review pressure monitoring, instrument calibration
Sampling manifold leak	5	Rupture of temporary containment	Revise training and operating procedure, train staff
Flexible line leak	1	Steam condensation/ galvanic corrosion	Remove lagging
Feed system valve leak	1	Formation of deposits	Reposition and inspect valves
Cylinder packing nut cracking	several	Stress corrosion cracking	Redesign and replace packing nuts
Cylinder valve leak	2	Mechanical impact	Revise procedures
Inadvertent UF_6 venting	1	Operating error	---

11.3.1.3 Summary of Prior NRC Evaluations of UF_6 Release Scenarios

Commercial UF_6 production facilities have operated at Metropolis, IL, and Gore, OK; government UF_6 diffusion separation facilities have operated at Portsmouth, OH; Oak Ridge, TN; and Paducah, KY; and fuel fabrication facilities have operated at a number of locations in the United States. In recognition of the potential hazards in handling UF_6, the NRC commissioned a study (Siman-Tov, 1984) which surveyed operating histories and synthesized a set of potential accident scenarios related to handling UF_6, as shown in Table 11.4. Because of differences in equipment and operations, the list includes some events which could not occur at the CEC.

The list of potential scenarios was reviewed to establish relevance of selected scenarios to the CEC. Relevant cylinder-related accidents include overheating of filled cylinders, failure of connectors and valves, and mechanical impact events; but exclude accidents involving the movement of liquid-filled cylinders. By virtue of the CEC design, UF_6 is present in the liquid state only while inside autoclaves. Cylinders containing liquid UF_6 are not moved at the CEC. With the exception of venting through a hydrolyzer, each of the potential process system failures could also occur at the CEC. Potential operator errors identified are also

Table 11.4 UF_6 accident scenarios (ORNL, 1984)

1.	UF_6 cylinder failures
1.1	Introduction of reactive hydrocarbons into a cylinder
1.2	Impact of a liquid-filled cylinder against an object or impact of an object on a cylinder
1.3	Valve or pigtail failure because of movement of a connected cylinder containing UF_6
1.4	Hydraulic rupture of a cylinder exposed to fire
1.5	Hydraulic rupture of an overheated cylinder
1.6	Hydraulic rupture of an overfilled cylinder
1.7	Heating or filling a defective cylinder
1.8	Heating a cylinder containing excessive volatile and/or gaseous contaminants
1.9	Dropping a liquid-filled cylinder
2.	UF_6 process system failures
2.1	Excessive heating of process equipment containing solidified UF_6
2.2	Fatigue failure of a process system
2.3	Impact of an object on a process system containing UF_6
2.4	Valve failure of a cylinder or a system containing UF_6
2.5	Pigtail failure
2.6	Process system loss of containment caused by natural phenomena
2.7	Heating a cold trap containing excessive volatile or gaseous contaminants
2.8	Heating an overfilled cold trap
2.9	Overheating a cold trap
2.10	Cold trap failure caused by corrosion, fatigue, or thermal shock
2.11	Venting of UF_6 through a hydrolyzer
3.	Nuclear criticality event
3.1	Nuclear criticality in a UF_6 vaporizer
3.2	Nuclear criticality resulting from a safe spacing violation
4.	Operator error
4.1	Valving a cold trap to a vacant position
4.2	Bypassing safety controls
4.3	Removing a valve from a cylinder containing UF_6

possible at the CEC. Quantities of UF_6 which might be released in these scenarios range from grams to metric tons.

11.3.1.4 Selection of Representative Accident Scenarios

The preceding analysis identified various potential events which could result in release of hazardous material from the CEC. The primary element of this analysis, a rigorous specific review of CEC equipment and operations was supplemented by a review of Urenco operating experience and prior NRC safety analyses. The resulting set of accident scenarios bounds the potential health and safety impacts of CEC operation. The analysis did not identify offsite events which could initiate releases of hazardous material from the CEC or CEC facility events which could affect operation of offsite public or industrial facilities. The primary potential hazard of operation of the CEC was found to be release of UF_6 which could threaten the health and safety of CEC workers and offsite individuals. In order to evaluate the nature of this potential hazard, a subset of potential accident scenarios intended to encompass the range of possible accidents was selected for detailed evaluation. The list of selected scenarios is summarized in Table 11.5. A generic criticality accident was included on the list in accordance with RG 3.34.

Table 11.5 Accident scenarios representative of potential CEC events

- Generic criticality
- Plant feed header line rupture
- Failure to de-gas line
- Flexible pipe leak in desublimer transfer
- Chemical storage area fire (inside building)
- Storage area fire/cylinder rupture
- Storage area collision/cylinder leak.

11.3.2 Description and Evaluation of Accidents

11.3.2.1 Criticality

Postulated occurrence of a criticality may be used to evaluate the adequacy of CEC activities in relation to public health and safety. The procedures described in RG 3.34, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Uranium Fuel Fabrication Plant" (NRC, 1979), are used for this purpose. Enriched uranium is present in the product, sampling, blending, decontamination, and waste management areas of the Separations Building. Because product take-off, blending, and sampling do not involve contact or effective interaction of moderators with uranic materials, this accident evaluation focused on the equipment decontamination process conducted in the Technical Services Area (TSA). In this process, citric acid solutions in tanks are used to remove uranium from equipment such as pumps and valves. Two tanks are used for decontamination operations, one with a volume of 100 liters, the other with a volume of 450 liters. Mass controls are used to ensure that concentrations of uranium in the tanks are well below critical limits. Total annual throughput of uranium is estimated to be 34 kilograms, and 35.5 kilograms is identified as the minimum amount required for a criticality event to occur. This analysis assumes that administrative and sampling controls have failed and that a criticality occurs in the 450-liter decontamination tank. The NRC staff reviewed the basis for the RG 3.34 guidance (Stratton, 1989) and identified two historical criticality incidents involving aqueous solutions of UF_6. In each case, the uranium concentration was greater than 50 times the administrative limits set for these tanks at the CEC, and the estimated number of fissions was less than 2.0×10^{17}. In a UO_2F_2-solution criticality accident, the number of fissions would be limited by the occurrence of a neutron absorption reaction involving fluorine atoms. However, in order to provide a conservative analysis for the CEC, a value of 5.0×10^{17} fissions was selected as the basis for analysis.

A large quantity of fission energy (about 5 kilowatts) would be released along with quantities of noble gases, fission products, and heavier radionuclides. Prompt gamma and neutron doses would also occur. Workers located in close proximity to a criticality event could incur serious health effects or fatalities. The event is assumed to occur in the decontamination area of the TSA, and the maximally exposed member of the public for prompt dose is assumed to be located at the fenceline for a period of 2 hours. Additional external dose would primarily result from exposure to noble gases, and the internal dose would primarily result from exposure to radioiodine. A puff dispersion model was applied for estimation of airborne radionuclide concentrations, consistent with the RG 3.34 description of the accident as a series of pulse events. For elevated releases, the largest atmospheric concentration per unit source (χ/Q) which would be exceeded 5 per cent of the time would occur at a distance of 400 meters. Analysis results are summarized in Table 11.6. Prompt doses are higher for the close-in location, but external and internal doses are higher for the more distant location. For either location, the doses are below the EPA Protective Action Guidelines for evacuation.

Table 11.6 Doses for hypothetical criticality accident (Sv)

Type of Exposure	Receptor at 165 m	Receptor at 400 m
Prompt gamma	3.0×10^{-4}	6.7×10^{-5}
Prompt neutron	8.1×10^{-4}	1.2×10^{-4}
External	6.9×10^{-6}	3.8×10^{-5}
Internal EDE	2.4×10^{-7}	1.7×10^{-6}
Total	1.1×10^{-3}	2.3×10^{-4}

11.3.2.2 Plant Unit Feed Header Rupture

CEC process piping is fabricated from an aluminum alloy and operates outside the autoclaves at sub-atmospheric pressures. The piping systems have not been designated as Safety Class I and are assumed to fail on the occurrence of the design basis earthquake. Average UF_6 flowrates are low, with the highest throughput occurring in the main feed header for each plant unit. If the main plant header ruptures, air initially flows into the pipe until pressure inside the pipe reaches atmospheric pressure. Pressure monitors in the autoclave exit line are designed to respond to this condition by closing a valve and terminating flow. However, the controller and valve are not seismically qualified and are assumed to fail. Autoclave Class I systems monitor autoclave air space conditions and do not respond to the hypothetical failures. Gaseous UF_6 continues to flow from the pipe at a flowrate determined by the available driving force. The design flow of 50 grams per second (gm/s) at a pressure drop of 172,320 pascals (Pa) (25 psia) is estimated to fall to 32 gm/s at a driving force of 68,930 pascals (10 psia). The UF_6 Handling Area Heating, Ventilation, and Air Conditioning (HVAC) system is designed to shutdown in the event of a major release. In the absence of forced ventilation, natural drafts from prevailing winds could cause a ground-level leakpath release of UF_6-contai[illegible] from the facility. As described in SER Section 4.2, the NRC staff has estimated a leak rate of 0.014 cubic meters per second (m^3/s) (30 cfm) to be representative of conservative meteorological conditions. Given the density of UF_6 at the projected conditions and the absence of ventilation, dilution in the compartment area occurs before the leakpath release. The release rate of UF_6 from the facility under these circumstances is estimated to be 0.06 gm/s. Estimation of release rate does not account for expected deposition and collection of UF_6 powder within the facility. Under the design basis earthquake conditions assumed as the initiator for the accident, the Separations Building maintains structural integrity, and operators are available to respond to potential system failures. This analysis assumes that a survey of the facility and response to system failures occurs within 2 hours. This assumption is conservative because the response required to terminate the release, that is, shut-off power to the heaters, can be accomplished in less than a minute. Uranium intakes at the fenceline and 400-meter locations are estimated to be 0.67 and 0.15 milligrams, respectively. Concentrations of HF at the fenceline and 400-meter locations are estimated to be 0.09 and 0.02 milligrams per cubic meter (mg/m^3), respectively. If the HVAC system continues to function, the uranium would be released from the plant stack. Fence-line and 400-meter location intakes under these conditions are estimated to be 0.26 and 0.92 milligrams, respectively. Concentrations of HF at the fenceline and 400-meter locations are estimated to be 0.04 and 0.12 mg/m^3, respectively. All estimated intakes and concentrations are well below the NUREG-1391 criteria for onset of clinically observable effects resulting from exposure to uranium and HF.

11.3.2.3 Failure to De-gas Process Lines

Connection and disconnection of lines potentially containing UF_6 are normal in the course of CEC operations. In all cases, operating procedures specify that the lines are evacuated or de-gassed before disconnection. Past experience shows that procedures are occasionally misunderstood or improperly executed, with a resulting uncontrolled release of UF_6 into the process area. A typical release occurs when the pump and associated line are not properly evacuated before disconnection for vacuum pump maintenance. Under such conditions, 20 liters of UF_6 would be released into the building air. Temperature and pressure corresponding to tails take-off conditions are assumed to yield a gas density of 2890 grams per cubic meter (gm/m^3) and a release of approximately 65 grams of UF_6. A release of this magnitude would be through the stack, and uranium intakes based on conservative dispersion conditions are predicted to be $2.4x10^{-4}$ and $3.6x10^{-4}$ milligrams for the fenceline and 400-meter locations, respectively. Average concentrations of HF at the fenceline and 400-meter locations are estimated to be $1.5x10^{-3}$ and $1.0x10^{-3}$ mg/m^3, respectively. The predicted intakes and concentrations are small fractions of the NUREG-1391 guidelines. Surveys conducted after events of this type at Urenco facilities have reported negligible worker doses.

11.3.2.4 Flexible Pipe Leak in Desublimer Transfer

Feed purification desublimers containing solid UF_6 are emptied by indirect heating with Freon. The sublimed material is transferred to a cylinder in the purification cubicle. A flexible pipe which connects the take-up cylinder to the transfer line is of a type known to develop leaks because of the nature of its use. This scenario investigates the consequences of failure of this flexible pipe. The desublimer is assumed to be transferring UF_6 at design conditions reported to be 13 kilograms per hour at desublimer pressure of 82,700 pascals (12 psia). Gaseous UF_6 is assumed to leak from the pipe into the purification cubicle and from the cubicle into the UF_6 Handling Area air space. Alpha-in-air monitors in the UF_6 Handling Area are calibrated to sound an alarm at levels equivalent to 3 parts per million (ppm) of HF. Operators would respond and investigate the cause for the alarm. At the specified release rate, the alarm level would be reached in approximately 5 minutes. This analysis assumes that the leak continues for 30 minutes until terminated by operator action. If the ventilation system is shut down, a leakpath release occurs, and the fenceline and 400-meter location uranium intakes are $4.7x10^{-3}$ and $1.0x10^{-3}$ milligrams of uranium, respectively. Concentrations of HF at the fenceline and 400-meter locations are estimated to be $2.6x10^{-3}$ and $5.4x10^{-4}$ mg/m^3, respectively. If the ventilation system continues to function, the fenceline and 400-meter uranium intakes are $7.4x10^{-3}$ and $2.6x10^{-2}$ milligrams, respectively. Concentrations of HF at the fenceline and 400-meter locations are estimated to be $4.0x10^{-3}$ and $1.4x10^{-2}$ mg/m^3, respectively. In all cases, the intakes and concentrations are small fractions of the NUREG-1391 protection guidelines. Mitigating factors not credited in the analysis are removal of uranium in the purification cubicle water spray and deposition of uranium in transport.

11.3.2.5 Chemical Storage Area Fire

Potentially toxic chemicals are stored or used in several areas of the Separations Building. The largest combination of potentially hazardous materials is stored in the Chemical Storage Area of the TSA. The hazard audit identifies combustible material (acetylene, petroleum oil, and carbon) and potentially toxic materials (carbon tetrachloride, bleach, and Freon-TF). The accident scenario selected to evaluate the hazard potential involving materials other than UF_6 is fire in the Chemical Storage Area. The accident is assumed to begin with an acetylene leak from the storage cylinder; the acetylene ignites on contact with an ignition source such as the forklift used in the area. The fire spreads to the petroleum oil and activated carbon, which are also consumed in the fire. The total energy release of approximately 18,460 megajoules (MJ) ($17.5 \times 10^{+6}$ BTU) is not large enough to spread to other areas of the building but is large enough to volatilize chlorine from carbon tetrachloride and bleach, and to evaporate Freon-TF. Spread of the fire is limited by the relatively low rate of heat release from the activated carbon and transfer of energy through the walls. Release of these potentially toxic chemicals constitutes the major hazard of this accident scenario. Combining the liquid bleach, assumed to be saturated sodium hypochlorite, and carbon tetrachloride yields an estimate of total potential chlorine release of approximately 65 kilograms. The entire inventory of Freon-TF (1,300 kilograms) is assumed to be evaporated and released. Published burning rates of flammable materials (Babrauskas, 1982) and the limited inventory of Freon-TF indicate that the minimum time for evaporation would be of short duration, on the order of minutes; thus, the chemical release is modeled as instantaneous. The scenario assumes, on the basis of joint frequency of meteorological data, dispersion estimates which would be exceeded 5 per cent of the time. Because the scenario does not involve UF_6, the ventilation system would continue to ventilate the room involved in the accident. Concentrations of chlorine at the fenceline and 400-meter locations are estimated to be 4.7 and 3.0 mg/m^3, respectively. Concentrations of Freon-TF at the fenceline and 400-meter locations are estimated to be 92 and 59 mg/m^3, respectively. Exposure times at the fenceline and 400-meter locations are estimated to be 2.5 and 6.0 minutes, respectively. The exposure time estimate was calculated as the time required for 95 percent of the chemical in the puff to pass the receptor location. The concentration level estimated for the onset of clinically observable effects with a 1-hour exposure (EPA ERPG-1) of chlorine is 1 mg/m^3. The American Conference of Governmental Industrial Hygienists (ACGIH) time-weighted average (TWA) and short-term exposure limits (STEL) for chlorine are 1 and 9 mg/m^3 (ACGIH, 1986), respectively. For Freon-TF, the TWA is 5,600 mg/m^3, and the STEL is 9,500 mg/m^3 (ACGIH, 1986). Although the estimated chlorine concentration levels are in the range of the guidance criteria, the exposure times are a small fraction of the criteria exposure times, and the impacts of the hypothetical exposure would be transitory and mild. The estimated levels of Freon-TF are small fractions of the guidance levels, and impacts of the hypothetical exposures would be negligible. Mitigating factors not included in the evaluation include operation of the fire suppression system, response of plant staff, and dilution of the chemicals in ventilation air before release.

11.3.2.6 UF_6 Storage Area Fire

Uncontrolled release from the liquid state poses the greatest hazard associated with handling UF_6. The magnitude of this hazard is evaluated by constructing a scenario involving overheating of a cylinder initially containing solid UF_6. The design of the CEC allows handling of UF_6 in the liquid state only in the Separations Building autoclaves. Overheating of cylinders in autoclaves is prevented by redundant Class I control systems. Consequently, analysis of overheating by immersion in a fire investigates other controls which may be applicable in preventing overheating of a cylinder. The scenario assumes that a UF_5 cylinder transporter is involved in a collision which ruptures the fuel tank. Fuel spills onto the ground and is ignited by the transporter engine. The fire engulfs the cylinder; the heat causes the cylinder to rupture and liquid UF_6 to spill out and flash into the vapor state. The NRC staff analyzed the consequences of this scenario in an earlier study of emergency response requirements for fuel cycle facilities (NRC, 1988b). The analysis estimated that a buoyant plume containing 7,100 kilograms of UF_6 is generated and converted by hydrolysis into a cloud containing 6,200 kilograms of uranyl fluoride and 1,620 kilograms of HF. Uranium intakes predicted for representative meteorological conditions are presented in Table 11.7.

Table 11.7 Uranium intake because of release from a ruptured cylinder (NRC, 1988b)

	Uranium Intake (milligrams)	
Distance (meters)	F Stability Class 1 m/s buoyant	D Stability Class 4.5 m/s buoyant
200	6	53
500	110	40
1,000	92	17
2,000	44	6
5,000	11	1.6
10,000	3	0.5

Intakes in excess of the NUREG-1391 guidance criteria are predicted for considerable distances from the release point.

A significant factor in the analysis is the assumption that heat generated in the fire is transferred to the cylinder and causes its rupture. However, this conservative assumption can be applied to derive a transporter fuel capacity limit which would prevent occurrence of the scenario. Research has established that representative temperatures for a pool fire approximate 800 °C (1,500 °F). Using this temperature, the NRC staff estimated heat transfer rates for conduction through the cylinder wall and forms of UF_6, and for radiative transfer of heat from the cylinder wall. The analysis established that maximum heat transfer rates are derived for radiative transfer from the cylinder wall to the UF_6. The minimum amount of energy required to produce a pressure excursion is estimated from the change of state of the UF_6. Initially, energy transferred to the UF_6 causes sublimation until the temperature and pressure reach the triple point. At these conditions, the UF_6 melts and expands as it transforms to the liquid state. Further addition of energy evaporates and expands the mixture of liquid and vapor UF_6 and thereby causes a rapid increase in pressure. On the basis of minimum heat required to liquefy UF_6 and the maximum heat transfer rate, a fire is estimated to take at least 20 minutes to rupture the cylinder. This estimate is similar to estimates derived by using finite element models (Clayton, 1991). The estimate is combined with empirically established burning rates to derive a minimum fuel requirement of approximately 280 liters (74 gallons). Limitation of transporter fuel inventories to less than this quantity, as stipulated in the PLC, will serve to prevent the occurrence of this release. The analysis incorporated conservative elements in order to provide a reasonable lower limit for the allowable transporter fuel inventory. The spilt fuel is assumed to pool under the cylinder even though the storage yard is covered with crushed rock and sloped in order to facilitate drainage, and the presence of workers operating the transporter is not credited for responding to the fire.

11.3.2.7 UF_6 Storage Area Collision

Straddle carriers and modified forklifts transport cylinders containing solid UF_6 between storage areas and the CEC buildings. Liquid UF_6 is present only in the Separations Building autoclaves, and cylinders containing liquid UF_6 are not transported. During transport, storage cylinders containing solid UF_6 have experienced minor damage such as small cracks or damage to the cylinder valve. One accident scenario investigates the consequences of an accident in which the cylinder valve is sheared by mechanical collision and thereby exposes solid UF_6 to the ambient atmosphere. Initially, the cylinder gas space contains UF_6 at sub-atmospheric pressure. After the valve is damaged, air enters the cylinder, and UF_6 begins to sublime and flow out of the cylinder. The rate at which the UF_6 sublimes is controlled by mass and energy transfer across a vapor film which develops around the solid UF_6. The film mass transfer process is driven by the difference in UF_6 partial pressure at the solid surface and in the air. The heat transfer process is driven by the difference in temperature at the solid surface and in the air. The rate of transfer for both processes is represented as equal to the product of the driving force and a transfer coefficient. The mass and heat transfer coefficients are estimated by using standard engineering correlations (Bird, Stewart, and Lightfoot, 1960). The mass and energy balances for the solid UF_6 and the UF_6 equilibrium partial pressure relation are solved simultaneously for the unknown mass transfer rate. On the basis of the area of a tails cylinder, a UF_6 release rate of 0.016 gm/s is estimated. For a release lasting two hours, the total UF_6 release is 115 grams. Actual response time would be less than two hours because the presence of a worker is an integral element of this scenario and fume leaks of UF_6 are readily contained. Uranium intakes at the fenceline and 400-meter locations are estimated to be 0.19 and 0.04 milligrams, respectively. Concentrations of HF at the fenceline and 400-meter locations are estimated to be 0.025 and 0.006 mg/m^3, respectively. Such uranium intakes and HF concentrations are well below the NUREG-1391 guidance criteria, and the consequences are expected to be minimal. Mitigating factors not considered in this analysis include prompt operator action to terminate the release and the equipment features (e.g. shrouds and stiffening rings) designed to prevent the occurrence of damage.

11.4 Summary of Potential Accident Analysis Impacts

The analyses described in this chapter consider a set of accident initiators which could lead to UF_6 releases of varying magnitude. The initiators included natural phenomena, operator error, and equipment failure. Analytical results indicate that releases because of operator error are small, with effects confined to the plant. Similarly, because of the relatively small inventories and process flows, equipment failure is found to have minor onsite and offsite effects. As shown by experience, the major potential threat to worker and public health and safety is release of UF_6 because of cylinder over-heating. The CEC plant design protects against this type of release by confining liquid UF_6 in autoclaves with redundant heater controls. Protection against a secondary potential cause of cylinder overheating, immersion in a fire, is implemented by limiting transporter fuel inventory. The NRC staff concludes that through the combined result of plant and process design, protective controls, and administrative controls, accidents at the CEC analyzed by the NRC staff do not pose an unreasonable risk to public health and safety.

12 QUALITY ASSURANCE

12.1 Background

Chapter 10 of the Louisiana Energy Services (LES) Safety Analysis Report (SAR) for the Claiborne Enrichment Center (CEC) (LES, 1993a) describes the quality assurance (QA) program applicable to the design, construction, start-up, and operation of the facility. In SAR Chapter 10, LES describes a graded, three-level QA program. QA Level 1 gives the greatest assurance of quality. It applies to System Class I items and activities, that is, those which prevent or mitigate events which could result in offsite exposures greater than the allowable limits defined in NUREG-1391 (NRC, 1991a). QA Level 2 provides an intermediate assurance of quality. It applies to System Class II items and activities, that is, those which might have some importance to safety but are not needed to prevent or mitigate events which could result in offsite exposures greater than the allowable limits defined in NUREG-1391. Normal industry practice is applied to the remaining items and activities, that is, those which have no safety importance.

The SAR indicates that, for System Class I items and activities, LES will follow the American Society of Mechanical engineers' (ASME) guidelines of the Basic Requirements, Supplements, and Appendices of ASME NQA-1 (ASME, 1989a). The SAR also indicates that, for these items and activities, LES will use the latest addenda of ASME NQA-2 (ASME, 1989b) as a guide to develop process procedures during the life of the CEC.

12.2 Findings

12.2.1 Design and Construction

The LES organization for the design and construction phases of the CEC is shown in Figure 12.1. The figure shows that the LES QA Director reports directly to the LES President, who has the overall responsibility for the QA program for the CEC and QA policies, goals, and objectives. The QA organization under the QA Manager, who reports to the QA Director, is responsible for establishing the documented QA program and verifying its effective implementation. As part of this responsibility, the QA Director ensures that the QA programs of the designer, contractors, and suppliers during the design and construction phases meet the applicable requirements of the LES QA program.

Also reporting to the LES President during the design and construction phases of the CEC is the Engineering and Contracts Manager, who is responsible for the CEC design, construction, and preparation for operation. The CEC Project Manager reports to the Engineering and Contracts Manager. The NRC staff concludes that these organization arrangements and assignments of responsibilities provide reasonable assurance of an acceptable QA program.

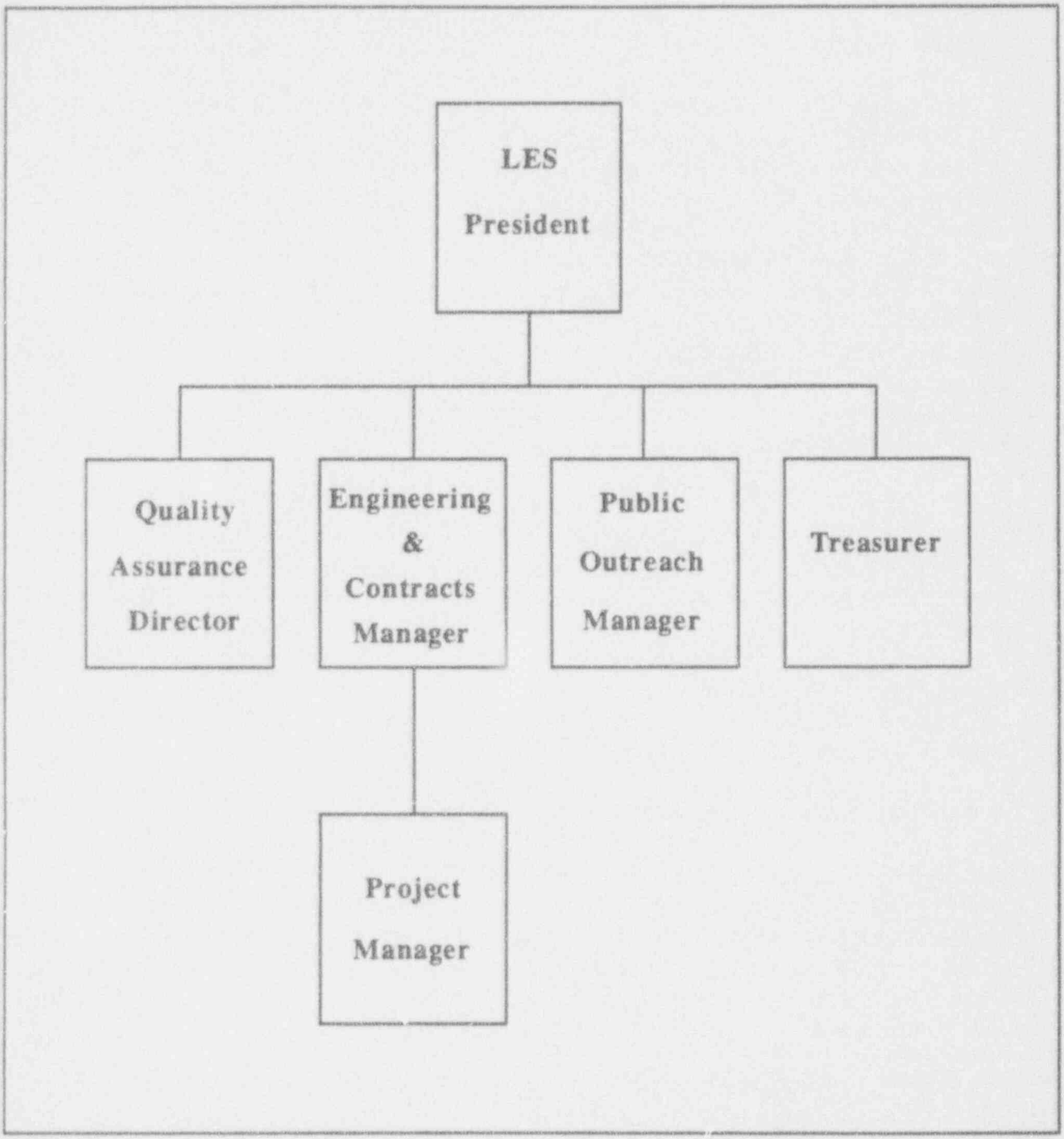

Figure 12.1 Louisiana Energy Services organization for design and construction

LES describes its QA program for design control for System Class I items in SAR Section 10.3. The description agrees with the design controls required by Criterion III of Appendix B of Part 50 of Title 10 of the Code of Federal Regulations (CFR) (NRC, 1956b). Other applicable criteria of 10 CFR Part 50, Appendix B also apply to System Class I items during the design phase. For example, SAR Section 10.5 addresses the controls and use of instructions, procedures, and drawings; 10.6, document control; 10.16, corrective action; 10.17, records; and 10.18, audits. The NRC staff concludes that the QA program for design control for System Class I items described in the SAR provides reasonable assurance of acceptable design.

The SAR indicates that, during the construction phase, LES will meet all eighteen criteria of 10 CFR Part 50, Appendix B for System Class I items and activities. In addition to the criteria noted above, SAR Chapter 10 addresses the remaining criteria of 10 CFR Part 50, Appendix B, such as procurement control, identification and control of items, control of special processes, inspection, tests, control of measuring and test equipment, and nonconformance control. The NRC staff concludes that the QA program for construction phase activities for System Class I items provides reasonable assurance of their acceptable construction.

LES describes its QA program for certain specified System Class II items and activities in SAR Section 10.19 . SAR Section 10.19 addresses organization, personnel qualifications, procedures, document control, design control, control of purchased items and services, control of processes and measuring and test equipment, inspections, nonconformances and corrective action, records, and audits. The NRC staff concludes that the QA program for System Class II items described in the SAR provides reasonable assurance of adequate controls for design and construction of these items.

12.2.2 Start-up and Operation

LES start-up and operating organization for the CEC is shown in Figure 12.2. The QA Director and the CEC Manager both report directly to the LES President. The QA organization is under the QA Manager, who reports directly to the CEC Manager and, as indicated by a dotted-line, indirectly to the QA Director. The NRC staff concludes that this organization arrangement provides reasonable assurance of an acceptable QA program.

The description of the QA program for the operations phase of the LES is the same as that given above for the design and construction phases. The NRC staff concludes that this QA program also provides reasonable assurance of adequate controls for plant start-up and operations.

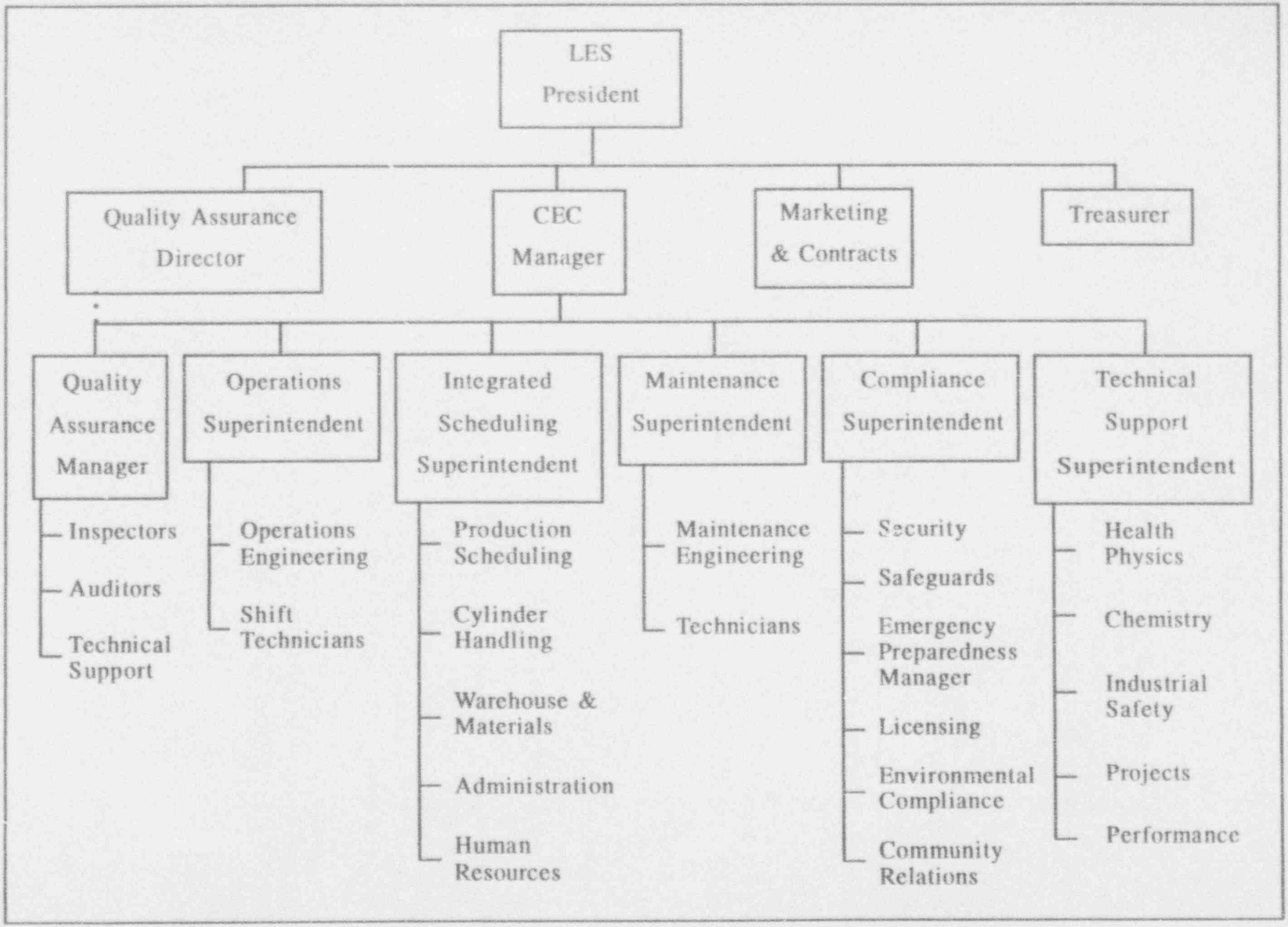

Figure 12.2 Louisiana Energy Services Claiborne Enrichment Center operating organization

12.3 Conclusion

The NRC staff has reviewed the LES QA program described in the CEC SAR (LES, 1993a) against the acceptance criteria given in "Quality Assurance during the Design and Construction Phases," (Section 17.1) and "Quality Assurance during the Operations Phase," (Section 17.2) of NUREG-0800 (NRC, 1987a). The NRC staff finds that the LES QA program, when implemented effectively, will meet the requirements of 10 CFR Part 50, Appendix B and thus concludes that the LES QA program is acceptable for the design, construction, start-up, and operation of the CEC.

13 FINANCIAL QUALIFICATIONS

13.1 Background

Four general and seven limited partners plan to construct, operate, and decommission a centrifuge uranium enrichment facility with ultimate production of 1.5 million separative work units (SWUs) per year. The partnership, Louisiana Energy Services (LES), plans to build this facility, the Claiborne Enrichment Center (CEC), in Claiborne Parish, Louisiana. Under Section 70.23(a)(5) of Title 10 of the Code of Federal Regulations (CFR) (NRC, 1956a), an applicant for a license under 10 CFR Part 70 should be "financially qualified to engage in the proposed activities in accordance with the regulations" in 10 CFR Part 70. In the absence of standards of financial qualifications for reviews specifically applicable to Part 70 license applicants, the NRC staff has used the guidance for newly formed entities contained in 10 CFR Part 50, Appendix C (NRC, 1956b). This analysis does not include findings or conclusions related to applicant's qualifications to decommission the facility. Decommissioning funding assurance is covered in Safety Evaluation Report (SER) Chapter 15. (Material in brackets has been deleted from the non-proprietary version of this analysis.)

13.2 Findings

13.2.1 Construction Costs

The applicant estimates that "hard" construction costs will be $816 million in 1992 dollars for the full planned capacity of 1.5 million SWU. For the first 0.5 million SWU of capacity, construction costs are estimated to be $313.5 million. Two incremental additions of capacity of 0.5 million SWU each are estimated to cost $251.8 million and $250.9 million, respectively. When interest, escalation, financing, and decommissioning costs are added to hard construction costs, the total project is estimated to cost [ ] million.

13.2.2 Sources of Funds

LES plans to fund the project with a mix of approximately [ ] million in debt, [ ] equity contributions by partners, and net revenues from enrichment services during the start-up phase. During the first increment of capacity, debt is projected to be [ ] million; partner equity contribution, [ ] million; and net revenues, [ ] million. LES's reliance on approximately 30 percent equity is positive because, by contrast, many analogous construction projects rely on 100 percent debt financing. Because of the partners' existing business relationships with banks and other lenders, debt financing for the initial increments of capacity will be financed without the assistance of financial intermediaries. Subsequent capacity additions will be financed either in a similar fashion or with the help of investment bankers on the basis of the status of the project at that time.

13.2.3 Contingency Funds

LES plans to meet contingencies for cost overruns and revenue shortfalls in several ways. Unforeseen construction contingencies will be minimized by the use of a turnkey contractor for the engineering, procurement, and construction of the facility. For cost overruns not covered under the turnkey provisions of the contract, LES will seek additional partner equity contributions. If cost overruns are much higher than anticipated, LES would cancel the project and leave an allowance for site stabilization.

The nature of the project means that there would be minimal potential public health and safety impact until its commencement and actual enrichment of UF_6 on site. LES indicates that operations would not begin until firm supply contracts with utility customers are in place. In addition, once construction funds have been expended, LES will have increasing incentive to complete the project because such costs are not recoverable except through sale of enrichment services. Operation and maintenance (O&M) costs are projected to be only a small fraction of operating income and so should not affect contingency planning. Additional contingencies would be covered through partner equity contributions to the extent not covered by contractors or insurance.

13.2.4 Financial Qualification

As indicated, LES ownership is vested with four general partners and seven limited partners. Partnership interests and capital contribution responsibilities are as follows (subject to rounding) :

General Partners	Percent
Urenco Investments	3.33
Claiborne Energy	2.37
Claiborne Fuels	0.88
Graystone Corp.	0.54
Limited Partners	**Percent**
Louisiana Power and Light	4.10
BNFL Enrichment	16.21
GnV	16.21
UCN Deelnemingen	16.21
Claiborne Energy	23.79
Le Paz	6.19
Micogen	10.16

Operating control is vested with the four general partners as follows:

Urenco Investments, Inc.	47%
Claiborne Energy Services, Inc.	33%
Claiborne Fuels, L.P.	12%
Graystone Corp.	8%

LES itself has had no reported income statements since its inception. Its most recent balance sheet indicates total capital of $16.8 million at the end of 1990. Its assets consist primarily of $16.8 million in deferred, start-up costs for the CEC.

1. Urenco Investments, Inc., is a wholly owned subsidiary of Urenco, Ltd., a United Kingdom company owned equally by International Nuclear Fuels plc (INFL), Ultra-Centrifuge Netherlands NV, and Uranit GmbH. In 1990, combined stockholder equity exceeded $1.37 billion, and combined net income exceeded $285 million, for these three companies. As a measure of liquidity, the combined ratio of current assets to current liabilities is 1.36, which is acceptable for this type of business. Urenco Investments, as a separate corporate entity, sustained a net loss in 1991 of $21,774 after a loss of $11,194 in 1990. Cash and cash equivalents on hand by the end of 1991 were $415 thousand. Urenco Investments and its owners have sufficient resources to make planned equity contributions and additional equity contributions within any reasonable range contemplated for the CEC.

2. Claiborne Energy Services, Inc., is a wholly owned subsidiary of Duke Power Company and did not submit separate financial statements. Shareholders' equity at the end of 1991 was approximately $4.1 billion. Duke realized net income of $583.6 million in 1991 after realizing $538.2 million in 1990. The ratio of current assets to current liabilities is 1.21, which is reasonable for a public utility. Cash flow in 1991 was $1.1 billion. Duke has sufficient resources to make planned equity contributions and additional equity contributions within any reasonable range contemplated for the CEC.

3. Claiborne Fuels, L.P., is a subsidiary of Claiborne Fuels, Inc., a wholly owned subsidiary of Fluor Daniel, Inc., in turn, a wholly-owned subsidiary of Fluor Corporation. Claiborne Fuels, L.P., did not submit separate financial statements. Shareholders' equity grew to $1.02 billion in 1991 from $864.0 million in 1990. Fluor realized net income of $160.8 million in 1991 after realizing $146.9 million in 1990. In 1991, Fluor's ratio of current assets to current liabilities was 1.37. Cash flow from operating activities was $229.7 million in 1991 after being $353.1 million in 1990. Fluor has sufficient resources to make planned equity contributions and additional equity contributions within any reasonable range contemplated for the CEC.

4. Graystone Corporation is a wholly owned subsidiary of Northern States Power Company. Total Graystone Corporation shareholder equity is [ ]. It lost [ ] for the year ended March 1992, a loss consistent with its budget for the same period. The ratio of current

assets to current liabilities is [ ], a ratio typical of a company sufficiently capitalized for its purposes but without income- or expense-producing activities. Northern States Power Company is an integrated electric utility operating primarily in Minnesota. Stockholders' equity at the end of 1991 was $1.58 billion. It realized net income of $224.0 million in 1991 after realizing $195.5 million in 1990. The ratio of current assets to current liabilities is 0.95, an acceptable ratio for an electric utility. Cash flow for 1991 was $444.7 million. Northern States Power has sufficient resources to make planned equity contributions and additional equity contributions within any reasonable range contemplated in the Claiborne application.

Six of the seven limited partners--BNFL Enrichment, GnV, UCN Deelnemingen, Claiborne Energy, Le Paz, and Micogen--are companies or subsidiaries of companies analyzed above. Only Louisiana Power and Light Company (LP&L), an integrated electric utility and wholly-owned subsidiary of Entergy Corporation, is without ties to the four general partners. Stockholders' equity for the period ending September 30, 1991, was $1.13 billion. For the 9 months ending September 30, 1991, LP&L realized net income of $151.9 million after realizing $150.4 million in the analogous period in 1990. Cash and cash equivalents for that period were $178.7 million. Entergy had net income in 1991 of $482.0 million and $478.3 million in 1990. Stockholders' equity was $4.23 billion, and cash and cash equivalents were $638.8 million at September 30, 1991. Thus, LP&L and Entergy Corp. have sufficient resources to make planned equity contributions and additional equity contributions within any reasonable range contemplated in the Claiborne application.

13.3 Liability Insurance

LES has committed to maintain nuclear liability insurance in the maximum commercially available amount of $200 million.

13.4 Conclusion

The applicant, LES and its partner owners, appears to be financially qualified to build and operate the proposed CEC. LES has identified sources of debt and equity capital for construction, and has reasonable assurance of securing them when needed. Operation costs are a small fraction of construction costs and are reasonably ensured by the enrichment services contracts being effected by LES.

The primary financial risk of the CEC project is the price in the domestic and international markets for enrichment services. LES market projections appear to take into account expected perturbations in enrichment prices because of fluctuations in worldwide demand and competitors' costs of supplying enrichment services. Planned CEC construction has several stages which make it easy to decide, on the basis of the estimated market for enrichment services, whether to proceed with or to cancel the project. Cancellation of the project at any stage before operation would not have an adverse effect on protection of public health and safety. As project construction progresses and expended construction costs become "sunk" costs, the decision to continue will depend on a comparison of future incremental construction and O&M costs to the expected revenues generated from enrichment services sales. Thus, because of high early construction costs, it becomes easier financially to justify continuation as construction proceeds unless major unplanned construction expenditures appear necessary. At the time operation commences, only variable O&M costs will be relevant to a decision to continue. The NRC staff concludes that the financial risk of the enrichment services market will diminish as construction proceeds and should not affect the protection of public health and safety.

14 SAFEGUARDS

14.1 Material Control and Accounting

14.1.1 Background

The NRC has established nuclear material control and accounting (MC&A) requirements for uranium enrichment facilities in Section 74.33 of Title 10 of the Code of Federal Regulations (CFR) (NRC, 1985b), "Material Control and Accounting of Special Nuclear Material". These requirements apply to uranium enrichment facilities authorized to produce and possess low-enriched uranium (LEU) of less than 10 weight percent (wt %) of the 235 atomic mass unit isotope of uranium (U-235). The nature of the operations and the types of materials at uranium enrichment facilities pose two unique problems addressed by NRC regulations. First, because the equipment used to enrich uranium to authorized enrichment levels can be used to produce higher enrichment levels, the NRC can not rule out the possibility of deliberate misuse of the equipment. Second, undeclared source material (SM) or LEU feed can be introduced into the process equipment for unauthorized production of enriched uranium. MC&A performance objectives to protect against, detect, and respond to such possibilities are established in 10 CFR 74.33. Other objectives encompassed in the regulations are consistent with MC&A requirements for other NRC-licensed facilities authorized to possess and use more than 1 effective kilogram of special nuclear material (SNM) of low strategic significance.

14.1.2 Current Requirements

NRC regulatory guidance documents used by the licensee to prepare the Fundamental Nuclear Material Control (FNMC) Plan and by the NRC staff to review and recommend final approval of the FNMC Plan are Regulatory Guide (RG) 5.67, "Materials Control and Accounting for Uranium Enrichment Facilities Authorized to Produce Special Nuclear Material of Low Strategic Significance," and NUREG/CR-5734, "Recommendations to the NRC on Acceptable Standard Format and Content for the Fundamental Nuclear Material Control (FNMC) Plan Required for Low-Enriched Uranium Enrichment Facilities" (Moran, 1991). These documents address commitments to be contained in the licensee's FNMC Plan which are necessary to meet the general performance objectives of 10 CFR 74.33(a).

Section 74.33(a) of Title 10 of the Code of Federal Regulations requires the licensee to establish, implement, and maintain an NRC-approved MC&A system capable of achieving the MC&A performance objectives of 10 CFR 74.33(a)(1) through (9):

- Maintain accurate, current, and reliable information of and periodically confirm the quantities and locations of source material and special nuclear material in the licensee's possession

- Protect against and detect production of uranium enriched to 10 wt % or more in the isotope U-235

- Protect against and detect unauthorized production of uranium of low strategic significance

- Resolve indications of missing uranium

- Resolve indications of production of uranium enriched to 10 wt % or more in the isotope U-235 (for centrifuge enrichment facilities this requirement does not apply to each cascade during its start-up process, not to exceed the first 24 hours)

- Resolve indications of unauthorized production of uranium of low strategic significance

- Provide information to the investigation of missing uranium

- Provide information to aid in the investigation of the production of uranium enriched to 10 wt % or more in the isotope U-235

- Provide information to aid in the investigation of unauthorized production of uranium of low strategic significance.

Section 74.33(b)(2) of Title 10 of the Code of Federal Regulations requires the licensee to implement the NRC-approved FNMC Plan before either receipt of more than a total of 5,000 grams of U-235 contained in natural, depleted, or enriched uranium, or the issuance of an NRC license to test or operate the enrichment facility.

An applicant's FNMC plan must demonstrate how the basic capabilities specified in 10 CFR 74.33(c) are achieved and maintained, and how such capabilities are used to achieve the performance objectives listed in 10 CFR 74.33(a). After accepting an FNMC plan and imposing it as a condition of license, the NRC judges the adequacy of a licensee's MC&A performance by inspecting for compliance with commitments and procedures described in the plan.

14.1.3 Required FNMC Plan Contents

The licensee must state in precise details the MC&A program that provides for:

- A management structure which ensures:
  - Clear overall responsibility for MC&A functions
  - Independence of MC&A management from production responsibilities
  - Separation of key MC&A responsibilities from each other
  - Use of approved written MC&A procedures
  - Periodic review of those procedures

- A measurement program which ensures that all quantities of SM and SNM in the accounting records are based on measured values

- A measurement control program which ensures:
  - Measurement biases are estimated, minimized through measurement control programs, and eliminated if statistically significant from inventory difference values of record
  - All MC&A measurement systems are controlled so that twice the standard error of the inventory difference (SEID), based on all measurement error contributions, is less than the greater of 5,000 grams of U-235 or 0.25 percent of the U-235 active inventory for each total plant material balance
  - Any measurements performed under contract are controlled so that the licensee can satisfy the SEID limitations stated above

- An inventory program which provides for:
  - Performing (unless otherwise required to satisfy 10 CFR Part 75) both a dynamic (nonshutdown) physical inventory of in-process uranium and U-235 at least every 65 days and a static physical inventory of all other uranium and total U-235 contained in enriched, normal, and depleted uranium located outside of the enrichment processing equipment at least every 370 calendar days (with static physical inventories being conducted in conjunction with a dynamic physical inventory of in-process uranium and U-235 so as to provide a total plant material balance at least every 370 calendar days)
  - Reconciling and adjusting the book inventory to the results of the static physical inventory and resolving or reporting an inability to resolve any inventory difference which is rejected by a statistical test with a 90-percent power of detecting a discrepancy of a quantity of U-235, established by NRC on a site-specific basis, within 60 days after the start of each static inventory

- A detection program, independent of production, which provides high assurance of detecting:
  - Production of uranium enriched to 10 percent or more in the U-235 isotope to the extent that SNM of moderate strategic significance could be produced within 370 calendar days
  - Production of uranium enriched to 20 percent or more in the U-235 isotope
  - Unauthorized production of uranium of low strategic significance

- An item control program which ensures:
  - Current knowledge is maintained for items with respect to identity, uranium and U-235 content, and stored location
  - Items are stored and handled, or subsequently measured, in a manner so that the amount of U-235 involved in any unauthorized removal of items, or uranium from items, greater than 500 grams of U-235 will be detected. Exempted from these provisions are licensee-identified items each containing fewer than 500 grams of

U-235 up to a cumulative total of 50 kilograms of U-235 and items that exist for fewer than 14 calendar days

- A resolution program that ensures the resolution of any shipper-receiver differences (SRDs) which are statistically significant and exceed 500 grams of U-235 on an individual batch basis and a total shipment basis for all SM and SNM

- An assessment program which:
  - Assesses independently the effectiveness of the MC&A system at least every 24 months
  - Documents the results of the above assessment
  - Documents management findings on the current effectiveness of the MC&A system
  - Documents any actions taken on recommendations from previous assessments.

The licensee must also supply an annex or appendix to the FNMC Plan which provides supplementary and general information about the facility and the MC&A system, for example, copies of blank record forms, site map, process diagrams, sample SEID calculations, and the like. The annex is not incorporated as a condition of license used as the basis for inspection. Procedures presented by the applicant or licensee to satisfy regulatory intent must be referenced in the plan itself rather than the annex and must provide adequate detail so as not to be largely dependent on examples or supplementary information in the annex for proper understanding.

In addition to following the commitments contained in the approved FNMC Plan, the licensee must establish records which comply with the record-keeping requirements of 10 CFR 74.33(d)(1), that is, records that demonstrate that the MC&A performance objectives, system features, and capabilities have been met. The records must be maintained in an auditable form, must be readily available for inspection, and must be retained for a minimum of 3 years. The licensee must maintain adequate safeguards against tampering with or loss of appropriate MC&A records.

14.1.4 Applicant's Proposed MC&A Programs

14.1.4.1 Applicant's FNMC Plan

The NRC staff has reviewed the FNMC Plan, Revision 3, dated October 1993, submitted by Louisiana Enrichment Services (LES), the applicant which seeks to build and operate the Claiborne Enrichment Center (CEC), and finds that it satisfies the performance objectives and system capabilities required by 10 CFR 74.33. However, the FNMC Plan still needs certain minor clarifications, prior to plant start-up, with respect to specific methodologies that are still undergoing development by the applicant.

14.1.4.2 Inspection

NRC staff will inspect the MC&A systems and controls as described in the facility FNMC Plan before process start-up. NRC staff will also review internal MC&A written procedures for adequacy before process start-up.

14.1.5 Conclusion

LES has provided commitments which meet the requirements of 10 CFR 74.33 by providing an acceptable FNMC Plan which describes acceptable methodologies for achieving the performance objectives of 10 CFR 74.33(a) and the system capabilities of 10 CFR 74.33(c). Included within such system capabilities are the means for precluding or detecting unauthorized enrichment activities. Thus, the NRC staff concludes that the LES FNMC plan, when implemented (with its compliance verified by the NRC staff), is acceptable for meeting the requirements of 10 CFR 74.33.

14.2 Physical Security

14.2.1 Current Requirements

The applicant must comply with the requirements of 10 CFR 73.67 (NRC, 1973b) for the possession of 10,000 grams or more of U-235 of SNM of low strategic significance. Specifically, the applicant must meet the performance objectives of 10 CFR 73.67(a), and the requirements for a physical security plan of 10 CFR 73.67(c), by complying with the measures for physical protection required by 10 CFR 73.67(f), and (g). Guidance is provided in RG 5.59, Revision 1, "Standard Format and Content for a Licensee Physical Security Plan for Protection of Special Nuclear Material of Moderate or Low Strategic Significance" (NRC, 1983b).

To achieve the level of physical security for SNM of low strategic significance at the facility which is necessary to meet the general performance objective, the applicant must comply with the following requirements of 10 CFR 73.67(f):

1. Store or use the material only within a controlled access area

2. Monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities

3. Assure that a watchman or off-site response force will respond to all unauthorized penetrations or activities

4. Establish and maintain response procedures for dealing with threats of thefts or thefts of this material.

To achieve the level of physical security with respect to transport of authorized SNM of low strategic significance which is necessary to meet the general performance objectives, the licensee must comply with the following requirements of 10 CFR 73.67(g):

1. Provide advance notification to the receiver of any planned shipments specifying the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier, and transport identification

2. Receive confirmation from the receiver before commencement of a planned shipment that the receiver will be ready to accept the shipment at the planned time and location and acknowledges the specified mode of transport

3. Transport the material in a tamper-indicating sealed container

4. Check the integrity of the containers and seals before shipment

5. Check the integrity of the containers and seals upon receipt of a shipment

6. Notify the shipper of receipt of the material as required in 10 CFR 70.54.

The applicant would be authorized to receive, process, use, store, and prepare for shipment authorized byproduct, SM, and SNM in accordance with 10 CFR Parts 19 (NRC, 1973), 20 (NRC, 1991e), 21 (NRC, 1977c), 30 (NRC, 1991d), 40 (NRC, 1961), 70 (NRC, 1956a), 71 (NRC, 1983c), 73 (NRC, 1973b), 74 (NRC, 1985b), and 75 (NRC, 1980a). For SNM, the shipper (typically a utility), not the applicant, is responsible for arranging for in-transit physical protection of the material, maintaining response procedures for dealing with threats of theft or thefts of this material, arranging for notification immediately upon arrival of a shipment at its destination or when a shipment is lost or unaccounted for after the estimated time of arrival at its destination, and initiating an immediate trace investigation and notifying the NRC Operations Center within 1 hour after recovery of or accounting for such lost shipment.

Before shipping any SNM, the applicant proposes to sign together with the shipper a document which specifies their respective responsibilities for physical protection of SNM while in-transit, in accordance with 10 CFR 73.67(g)(3).

Although the applicant is not the shipper, the applicant proposes to verify before the release of any SNM for shipment that the party responsible for in-transit physical protection requirements has arranged for in-transit physical protection of the material, has established and maintains response procedures for dealing with threats or attempted thefts of material, has made arrangements for notification immediately upon arrival of a shipment at its destination or when a shipment is lost or unaccounted for after the estimated time of arrival at its destination, and has made arrangements for initiating an immediate trace investigation and for notifying of the NRC Operations Center within one hour after recovery or accounting of such lost shipment (10 CFR 73.67(g)(3)).

14.2.2 Applicant's Proposal

The applicant has proposed the following measures to comply with the requirements of 10 CFR 73.67(f) and (g):

Storage and Use of Material

The applicant has proposed to enclose the entire facility within a controlled access area demarcated by a chain link fence. A clear zone surrounds the perimeter fencing. Access into the area is controlled and monitored by watchmen. Materials and processes are located in a separate building into which access is further controlled and monitored.

Access Controls

The personnel gates are continuously attended and monitored by security personnel. Vehicle gates are also attended and monitored by security personnel when in use. Access into the site is based upon established need and authorization. Uncleared visitors are escorted. Personnel and packages are searched for items which could be used for theft or sabotage. Access control procedures include search of packages and persons acting suspiciously. Access through the perimeter fencing is through personnel and vehicle gates. Access into the Separations Building, which contains the centrifuge cascades, is restricted to cleared persons who require such access to perform their duties.

Detection of Unauthorized Penetrations or Activities

The applicant has proposed to monitor controlled access areas and storage areas by security patrols. The security patrols are supplemented by closed-circuit television surveillance from an alarm station to assist in monitoring and assessment. The Separations Building is protected with intrusion alarms and additional closed-circuit television. Communication with the local law enforcement authority is maintained and checked on a periodic basis.

Response to Unauthorized Penetrations or Activities

The applicant has proposed a response to unauthorized penetrations or activities through security patrols and assessment capability, communications between the security patrols and the alarm station, and communications between the alarm station and the local law enforcement authority. The applicant has also proposed to develop a response to unauthorized penetrations or activities by establishing liaison with and providing site familiarization training to local law enforcement personnel. On-site security personnel are responsible for initial and immediate response and investigation of indications of suspicious activities or penetration attempts.

Personnel Trustworthiness

The applicant has proposed to require that all personnel working at the facility be screened for personnel trustworthiness. Personnel will be cleared at levels commensurate with their site access and duties. These measures are above those required by regulation.

Advance Notification and Confirmation of Shipment

The applicant has proposed procedures for advance notification before shipment and for confirmation after arrival at destination for each shipment of SNM.

Inspection of Shipments

The applicant has proposed use of containers, seals and locks during transport and inspections before and on receipt of shipment of SNM.

Verification of Response Procedures for Transport

The applicant has proposed to verify that arrangements for in-transit physical protection measures have been established and are maintained by the responsible party (shipper) to respond to a threat of theft or attempted thefts of material during transport; that arrangements have been made for immediate notification of arrival of a shipment, its loss or its being unaccounted for after the estimated time of arrival at its destination; and that arrangements have been made to conduct an immediate trace investigation of any shipment that is lost or unaccounted for, and to report such loss and recovery of the material to the NRC Operations Center.

14.2.3 Comparison of Proposal to Requirements

The applicant has complied with the requirements of 10 CFR 73.67(f)(1) by providing for the storage and use of low-enriched SNM only within a controlled access area.

The applicant has complied with the requirements of 10 CFR 73.67(f)(2) by providing for monitoring of the controlled access areas to detect unauthorized penetrations or activities.

The applicant has complied with the requirements of 10 CFR 73.67(f)(3) by establishing a response to unauthorized penetrations or activities by using security patrols, local law enforcement and a communication capability.

The applicant has complied with the requirements of 10 CFR 73.67(f)(4) by providing response procedures and maintaining those procedures and copies of superseded material for 3 years.

The applicant has complied with the requirements of 10 CFR 73.67(g)(1)(i), (ii), (iii), and (iv) by providing for advance notification before shipment, by confirmation before shipment that the receiver will be ready to accept the shipment, by use of a tamper-indicating and sealed container during transport, and by the inspection of containers and seals before shipment.

The applicant has complied with the requirements of 10 CFR 73.67(g)(2)(i) and (ii) by providing a check on the integrity of containers and seals on arrival of a shipment and by notifying the shipper of receipt of the material.

The applicant has complied with the requirements of 10 CFR 73.67(g)(1)(v), 73.76(g)(2)(iii), and 73.67(g)(3) by signing before shipment of SNM an agreement with the shipper which specifies their respective responsibilities for physical protection of SNM while in-transit. The applicant will also verify before shipment that response procedures have been established and are maintained by the party responsible for arrangements for physical protection of the shipment; verify that arrangements have been made for notification of the shipper immediately on arrival of the shipment at the destination, or for any such shipment lost or unaccounted for after the estimated time of arrival at its destination; and verify that arrangements have been made to conduct a trace investigation of any shipment lost or unaccounted for after the estimated time of arrival and to notify the NRC Operations Center within 1 hour after recovery of or accounting for such lost shipment in accordance with 10 CFR 73.71.

14.2.4 Preoperational Physical Security Inspection

Before the receipt of SNM the site will be subject to a preoperational physical security inspection to ensure that all physical security systems, components, and procedures are installed and implemented correctly and support the commitments in the physical security plan.

14.2.5 Conclusion

The NRC staff reviewed the applicant's Physical Security Plan, Revision 3, dated October 1993, and found it to satisfy the performance objectives and system capabilities required by 10 CFR 73.67. This plan, which provides details on the physical protection system, components, responsibilities and procedures is not publicly releasable.

The applicant has made commitments which meet the requirements of 10 CFR 73.67 by providing an acceptable physical security plan which protects against theft by specifying measures for early detection and assessment of unauthorized access or activities within the controlled access area which contains SNM, for early detection of removal of SNM by an external adversary from the controlled access area, for proper placement and transfer of custody of SNM, and for response to indications of an unauthorized removal of SNM and notification of the appropriate response forces to facilitate its recovery.

Accordingly, the NRC staff concludes that the physical security plan for the LES' CEC, when implemented and verified by the Commission, is acceptable in meeting the requirements of 10 CFR 73.67 (a),(c),(f), and (g).

15 DECONTAMINATION AND DECOMMISSIONING

15.1 Background

Facilities licensed by the NRC under Parts 40 and 70 of Title 10 of the Code of Federal Regulations (CFR) (NRC, 1956a and NRC, 1961), are decommissioned by licensees in order to permit release of the site and facilities for unrestricted use and to terminate the license. Licensees are required to demonstrate that the premises, where licensed activities were conducted, are suitable for release for unrestricted use. As a first step, licensees generally characterize the facilities and site where licensed activities were conducted and which could have been contaminated as a result of licensed activities, and report the results to the NRC. Depending on the type of licensed activities conducted at the facility, licensees may demonstrate adequacy for unrestricted release in some other manner.

Site characterization generally includes determination of:

- direct gamma radiation levels at one meter from surfaces in terms of microrads per hour or equivalent
- beta and direct gamma radiation levels at one centimeter from surfaces in terms of microrads per hour or equivalent
- removable and fixed alpha radiation surface concentrations in terms of disintegrations per minute per 100 square centimeters or equivalent
- radionuclide, gross alpha, and gross beta volume concentrations in water in terms of microcuries per milliliter or equivalent, and in solids such as soils and concrete, in terms of picocuries per gram or equivalent.

Prior to decommissioning, licensees must submit a detailed decommissioning plan to the NRC for approval, if it is required by license condition, or if a condition contained in 10 CFR 40.42(c)(2) is met.

The decommissioning plan generally includes:

- a description of the proposed decontamination and decommissioning (D&D) activities
- a description of the methods used to protect workers and the environment during D&D
- a description of the planned final radiation survey to be conducted by the licensee to demonstrate that the facility has been decontaminated and can be released for unrestricted use

- a cost estimate for decommissioning, and if this estimate is higher than the funds set aside, then a plan for assuring the availability of additional funds required to complete decommissioning.

The NRC will approve the decommissioning plan if the plan demonstrates that decommissioning will be completed in a timely manner, and that the health and safety of workers and the public will be adequately protected, both during and after decommissioning activities. After decommissioning activities are completed, the licensee generally conducts a final radiation survey and submits the results to the NRC. The NRC generally performs a confirmatory survey to verify the results of the licensees's final radiation survey. If the NRC determines that data obtained from the licensee's final radiation survey or the NRC's confirmatory survey does not adequately demonstrate that the premises are suitable for release for unrestricted use, the NRC informs the licensee of appropriate further actions required for terminating the license.

15.2 Applicant and NRC Staff Analyses and Conclusions

In Section 11.8 of the Claiborne Enrichment Center (CEC) Safety Analysis Report (SAR) (LES, 1993a), the applicant has identified activities required for decommissioning and estimated decommissioning costs. Activities and costs are based on actual operating experience. Urenco has a fully operational dismantling and decontamination facility at its Almelo plant located in the Netherlands; the applicant used data and experience from this operating facility to estimate decommissioning activities and costs. The applicant has committed to making financial arrangements to cover all costs required for returning the site and facilities to a condition suitable for unrestricted use. The applicant has also committed to updating the decommissioning cost and funding estimates at least once every five years. The cost estimate at the time of decommissioning will be based on a more detailed CEC plan for completion of decommissioning which will be submitted to the NRC for approval.

The following subsections conceptually describe decommissioning plans, cost estimates, and funding arrangements, and discuss the decontamination aspects of the program as described by the applicant. These subsections also contain the NRC staff's assessment of the applicant's proposed conceptual decommissioning plan, cost estimate, and funding plan.

15.2.1 Conceptual Decontamination and Decommissioning Plan

The plan for decommissioning is to promptly decontaminate or remove all materials from the site which prevent release of the facility for unrestricted use. This approach, referred to in the industry as DECON, avoids long-term storage and monitoring of wastes on site. For this reason it is the preferred alternative for decommissioning. The other industry methods, SAFSTOR and ENTOMB, require onsite storage and monitoring of wastes. The applicant states in Section 11.8.1 of the SAR that the types and amounts of wastes that will be produced at the CEC will not warrant delays in waste removal. The staff agrees that the

DECON approach for decommissioning the CEC is the preferred alternative over SAFSTOR and ENTOMB.

This subsection describes the applicant's approach, as well as the NRC staff's assessment. Further details are contained in SAR Section 18.1.1.

15.2.1.1 Decommissioning Design Features

The applicant states in SAR Section 11.8.1.1 that specific features will be incorporated into the facility design which will facilitate D&D of the facility. The major features are discussed below.

Radioactive Contamination Control

The applicant states in SAR Section 11.8.1.1.1 that the following features will serve to minimize the spread of radioactive contamination during operation, and therefore simplify eventual plant decommissioning. As a result, worker exposure to radiation, and radioactive waste volumes will be minimized as well.

- Certain activities during normal operation are expected to result in surface and airborne radioactive contamination. Specially designed rooms will be provided for these activities to preclude contamination spread. These rooms will be isolated from other areas and will be provided with ventilation and filtration. The Pump Disassembly Room and the Contaminated Workroom meet these specific design requirements. See Figure 15.1, for room locations. Figure 15.1 also shows locations of areas likely to have controlled access, for occupational safety reasons, during the operating life of the plant.

- All areas of the plant will be sectioned off into clean areas and potentially contaminated areas. The potentially contaminated areas will be called Radiation Control Areas (RCAs) and will have access control requirements. Areas more likely to be contaminated will be called Radiation Control Zones (RCZs). These RCZs will have additional access controls, and a number of requirements will be imposed on work procedures for contamination control. As discussed in SER Section 8.4.1.4, clean-up criteria for surface contamination in RCZs are 5,000 dpm/100 cm^2 removable alpha or beta/gamma and 250,000 dpm/100 cm^2 fixed alpha or beta/gamma. See Figure 15.1, for a floor plan showing the RCA/RCZ boundaries. See SER Section 8.4 for definitions of RCAs and RCZs. All procedures for these areas will fall under the health physics program, and will be designed to minimize the spread of contamination and simplify eventual decommissioning.

- A minimum number of non-radioactive process equipment and systems will be used in locations subject to probable contamination. This measure limits the size of the RCZs, and limits the activities occurring inside these areas.

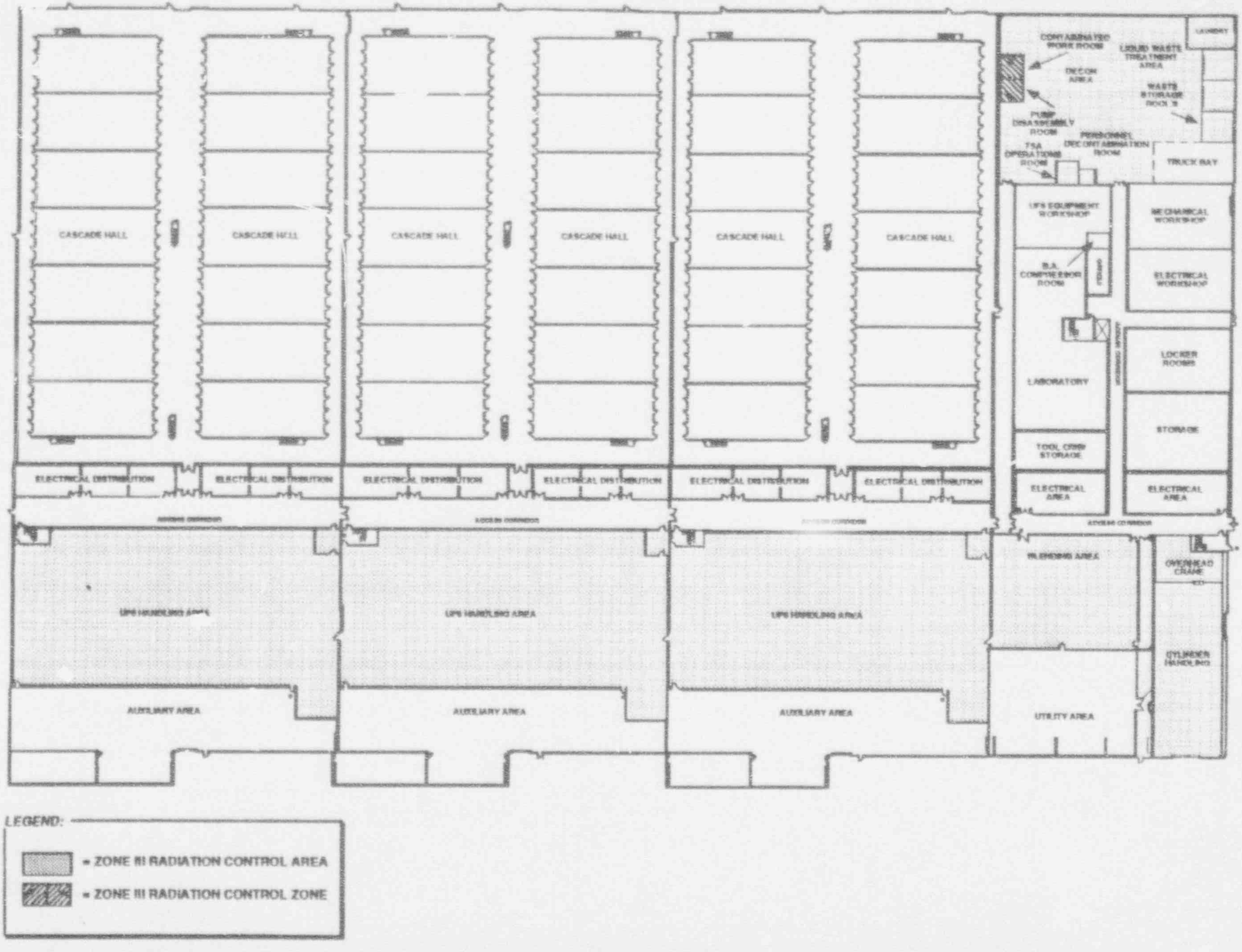

Figure 15.1 Claiborne Enrichment Center radiological access zones (LES, 1993a)

- Local air filtration will be provided for areas with potential airborne contamination to preclude its spread. Portable ventilation units and fume hoods will filter contaminated air in these areas.

- Curbing, pits, or other barriers will be provided around tanks and components which contain radioactive wastes. These will serve to control contamination spread in case of a spill.

The NRC staff agrees with the applicant that applying the features mentioned above will minimize spread of contamination. Therefore, eventual plant decommissioning will be simpler than if these features were not incorporated.

<u>Worker Exposure and Waste Volume Control</u>

The applicant states in Section 11.8.1.1.2 of the SAR, that the following features will serve to minimize worker exposure to radiation, radioactive waste volumes during decontamination activities, and the spread of contamination.

- During construction, a washable epoxy coating will be applied to floors and walls that might be radioactively contaminated during operation. The washable coating will simplify decontamination activities and lower waste volume during the decontamination process. The coating will be applied to all floors and walls in areas controlled for radiation. See Figure 15.1.

- To reduce waste volume during decommissioning, sealed nonporous pipe insulation will be used in areas likely to be contaminated.

- To minimize the time of worker exposure, ample access will be provided for dismantling and removal of equipment that may be contaminated.

- Tanks will have accesses for entry and decontamination. Design provisions will also be made to allow complete draining of the wastes contained in the tanks.

- Connections in the process systems will be provided for thorough purging at plant shutdown. This will allow removal of a significant portion of radioactive contamination prior to disassembly.

- Design drawings, to be produced for all areas of the plant, will simplify the planning and implementing of decontamination procedures. This in turn will shorten the durations of worker exposure to radiation.

- Worker access to contaminated areas will be controlled to assure that workers wear proper protective equipment and limit their time in the areas.

The NRC staff agrees with the applicant that the features provided above will minimize worker exposure and waste volumes during decontamination activities.

15.2.1.2 Decommissioning Steps

The applicant has briefly described the decommissioning methodology to be employed at the CEC. The applicant states (LES, 1993a) that implementation of the DECON alternative for decommissioning the CEC may begin immediately following final plant shutdown. The applicant estimates that the DECON alternative will take approximately seven years from plant shutdown to completion of the final radiation survey. The order of activities to support decommissioning will be: (1) characterization of the CEC facility and site, (2) generation of a detailed decommissioning plan, (3) installation of decontamination facilities, (4) process system purging, (5) equipment dismantling and removal, (6) decontamination of equipment and the facility, (7) destruction of Confidential and Secret Restricted Data material, (8) sale of salvage, (9) disposal of wastes (10) completion of a final radiation survey, and (11) completion of NRC's confirmatory survey. The applicant's overview and explanation of the major steps are discussed below in more detail.

Overview

Decommissioning, using the DECON approach, requires residual radioactivity to be reduced below specified levels so the facilities may be released for unrestricted use. Current NRC guidelines for unrestricted release serve as the basis for decontamination costs estimated by the applicant. The applicant states in Section 11.8.1.2.1 of the SAR that portions of the facility which do not exceed decommissioning contamination limits may remain as is. The applicant intends to remove all enrichment-related equipment from the buildings in such a manner that only the building shells and site infrastructure remain. The equipment to be removed by the applicant will include: all piping and components from systems providing UF_6 containment, systems in direct support of enrichment (such as refrigerant and chilled water), radioactive and hazardous waste handling systems, contaminated HVAC filtration systems, etc. The remaining site infrastructure will include services such as: electrical power supply, treated water, fire protection, HVAC, plant cooling water, communications, and sewage treatment.

The applicant will install two new facilities dedicated for decontamination of plant components and structures. Existing plant buildings are assumed to house the facilities. One facility will be especially designed to accommodate repetitive cleaning of thousands of centrifuges, and the other will serve as a general purpose facility used primarily for larger components. The two new facilities will be the primary locations for decontamination activities. The small decontamination area in the Separations Building TSA, used during normal operation, may also handle small items at decommissioning.

The NRC staff is in agreement with the applicant's proposal to remove all enrichment-related equipment from the buildings at the time of decommissioning, and leave only the uncontaminated building shells and site infrastructure. The NRC staff also finds acceptable the applicant's concept of installing two new facilities dedicated for decontamination of plant components and structures; using one for repetitive cleaning of centrifuges, and the other as a general purpose facility primarily for larger components. It should be noted that at the time of decommissioning, detailed descriptions of the two new facilities and the D&D procedures will be reviewed by NRC staff prior to authorizing D&D activities not authorized under existing regulations or the operating license.

The applicant states in Section 11.8.1.2.1 of the SAR that decontaminated components may be reused or sold as scrap. All equipment that is to be reused or sold as scrap will be decontaminated by the applicant to a level at which further use is unrestricted. Table 15.1 lists major items on the site expected to require decontamination. Materials which cannot be decontaminated will be disposed of by the applicant in a licensed radioactive waste disposal facility. At the time of decommissioning, the applicant has committed to selling or converting the depleted UF_6 (DUF_6) to a stable non-volatile uranium compound, such as U_3O_8, and to disposing the material offsite, in accordance with regulatory requirements. It should be noted that the applicant is committed to a DUF_6 possession limit of 80,000 tonnes or a lower quantity produced during 15 years of CEC operation. Specifically, no cylinder filled with DUF_6 will be stored onsite longer than 15 years. Thus, the NRC staff expects a significant fraction of the DUF_6 to be dispositioned before decommissioning which is expected to begin after about 30 years of plant operation.

The applicant states in Section 11.8.1.2.1 of the SAR that contaminated portions of the buildings will be decontaminated as required. According to the applicant, structural contamination should be limited to the areas indicated on Figure 15.1 as being inside the Radiation Control Zones of the plant. The remainder of the site, including the holding pond and all land area, is not expected to require significant decontamination. The applicant states in Section 11.8.1.2.1 of the SAR that good housekeeping practices during normal operation will maintain the other areas clean. When decontamination is complete, areas and facilities on the site will be surveyed by the applicant to verify that further decontamination is not required. The applicant will continue decontamination activities until the entire site is demonstrated to be suitable for unrestricted use. The NRC will independently confirm that the site is suitable to be released for unrestricted use.

The NRC will not authorize unrestricted release of materials and equipment unless all release criteria applicable at the time of decommissioning have been met. The NRC will not authorize release of the site for unrestricted use until the applicant adequately demonstrates that all decommissioning criteria applicable at the time of decommissioning have been met.

Table 15.1 Items for decontamination at decommissioning

Category	Description	Quantity
Pumps	Vent vacuum pumps	43
	Process vacuum pumps	198
	Waste disposal pumps	18
Centrifuges	Aluminum (tons)	5000
Piping	Aluminum, some steel (tons)	280
Gaseous effluent piping/ductwork	Diameter ≥ 14" (ft)	700
	Diameter 8 to 12" (ft)	500
	Diameter ≤ 6" (ft)	10,000
HVAC	TSA ductwork (length, ft)	400
	Filter housing (7'x7'x17')	3
Bldg surfaces	Floors and walls (ft^2)	10,000
Valves	Process valves	2500
Traps	Chemical traps	28
	Carbon traps	15
	Activated alumina traps	61
	Oil traps	49
	Sodium fluoride traps	42
Tanks	Liquid waste tanks	18
	Decontamination baths	4
Effluent pits	Plant unit and TSA pits	4
Other equipment	Desublimers	13
	Cont. dump surge vessels	42
	UF_6 sample rigs	6
	Clothes washer	1
	Clothes dryer	1

Table 15.1 Items for decontamination at decommissioning (continued)

Category	Description	Quantity
Other equipment (cont.)	Degreasing units	2
	Fomblin oil fume hood	1
	Fomblin oil centrifuge	1
	LWD dryer	1
	..yer feed filter	1
	LWD Precipitation Centrifuge	1
Stillage assembly	48" stillage	26
	30" stillage	26
Final decontamination facility	Centrifuge transporter	2-3
	Centrifuge manipulator	2-3
	Centrifuge dismantling equip. (table/saw/tank/box)	1
	Sawing machines	4
	Degreasers	2
	Decontamination tanks	6
	Wet blast cabinet	1
	Crusher	1
	Smelter	1

Decontamination Facility Construction

Two new facilities for decontamination will be installed by the applicant in existing plant buildings. The time for installation estimated by the applicant is approximately one year following plant shutdown. Details of the facilities, as provided by the applicant in Section 11.8.2 of the SAR, are contained in Section 15.2.1.8 of this document.

The NRC staff agrees with the applicant's concept of constructing decontamination facilities in the existing plant buildings. At this time, taking into consideration the limited description of the decontamination facilities provided by the applicant (see Section 15.2.1.8), the staff agrees with the applicant's estimate of one year for installing these facilities.

Process System Purging

The applicant states in Section 11.8.1.2.3 of the SAR that at the end of the useful life of the facility, the enrichment process will be shut down and UF_6 will be removed to the fullest extent possible by normal process operation. This will be followed by evacuation and purging with nitrogen. The applicant estimates that the shutdown and purging portions of the decommissioning process will take approximately three months.

The NRC staff agrees with the applicant's proposal and estimate of three months to remove the UF_6 remaining in the process system after plant shutdown.

Dismantling

Dismantling involves cutting out, disconnecting, etc., all components requiring removal. The operations themselves may be simple, but very labor intensive. Depending on the level of contamination, they may require the use of protective clothing. The applicant states in Section 11.8.1.2.4 of the SAR that the work process will be optimized, considering the following:

- Minimizing contamination spread and the need for protective clothing

- Balancing the number of cutting and removal operations with the resultant decontamination and disposal requirements

- Optimizing the rate of dismantling with the rate of decontamination facility throughput

- Providing storage and laydown space required, as impacted by retrievability, criticality safety, security, etc.

The applicant states in Section 11.8.1.2.4 of the SAR that the details of the complex optimization process will necessarily be decided near the end of plant life, taking into account specific contamination levels, and available waste disposal sites. To avoid laydown space and contamination problems, dismantling will likely be allowed to proceed no faster than the downstream decontamination process. The applicant estimates that dismantling and decontamination will take approximately three years. The decontamination process is addressed separately in detail in Section 15.2.1.8.

The NRC staff agrees with the applicant's estimate of three years to dismantle all components requiring removal. However, before NRC approval of a final decommissioning plan, only those dismantling operations which are within the criteria established under 10 CFR 70.38(c)(2) may be conducted.

Sale/Salvage

Items to be removed from facilities such as the CEC can be categorized as potentially re-usable equipment, recoverable scrap, and wastes. However, based on a 30-year operating life, the applicant does not assume that the facility operating equipment has any reuse value (See Section 11.8.1.2.6 of the SAR). This includes such equipment as diesel generators which are unlikely to be used throughout the operating life of the plant. According to the applicant, wastes will also have no salvage value.

With respect to scrap, the applicant claims in Section 11.8.1.2.6 of the SAR that a significant amount of aluminum will be recovered, along with smaller amounts of steel, copper, and other metals. For security and convenience, these materials will likely be smelted to standard ingots, then sold at market price. However, the applicant has not assigned salvage value to scrap in estimating their decommissioning funding requirements. See SER Section 15.2.2 and SAR Section 11.8.1.2.6.

The applicant states in Section 11.8.1.2.6 of the SAR that aluminum was reclaimed by Urenco from the decommissioning of two pilot plants. Centrifuges and other equipment containing aluminum were dismantled, further cut up into small pieces, decontaminated, and sent offsite to a smelter. Of the aluminum delivered to the smelter, almost 90 percent was suitable for resale. The remaining slag was disposed of as non-radioactive waste. The aluminum for resale contained between 2 and 4 ppm uranium. The current allowable uranium in aluminum limit in Europe is equivalent to 5 ppm. For natural uranium, this limit in terms of radioactivity is less than 0.09 Bq/g (2 pCi/g). According to the applicant, the sale price of the aluminum has generally been between 75 and 85 percent of the European spot market price. In 1990, in The Netherlands, the price was approximately 2.5 guilders ($1.39) per kilogram of aluminum.

It is possible that a large portion of the decontaminated aluminum will be recyclable. However, consistent with NRC guidance, the applicant is not assigning salvage value to this material for the purpose of estimating funding requirements for decommissioning.

Disposal

The applicant states in Section 11.8.1.2.7 of the SAR that all wastes produced during decommissioning will be collected, handled, and disposed of in a manner similar to that described for those wastes produced during normal operation. According to the applicant, wastes will consist of normal industrial trash, non-hazardous chemicals and fluids, small amounts of hazardous materials, and radioactive wastes. The radioactive waste will primarily be crushed centrifuge rotors, trash, and citric cake. Citric cake will consist of uranium and metallic compounds precipitated from citric acid decontamination solutions. The applicant estimates approximately 100 cubic meters of radioactive waste to be generated over the seven-year period of facility decommissioning activities. This waste will be subject to further volume reduction processes prior to disposal.

The NRC staff agrees with the applicant's assessment of generating 100 cubic meters of low-level waste from D&D activities for the purposes of estimating low-level radioactive waste disposal costs.

The applicant states in Section 11.8.1.2.7 of the SAR that radioactive wastes will ultimately be disposed of in licensed low-level radioactive waste disposal facilities. Hazardous wastes will be disposed of in hazardous waste disposal facilities. Non-hazardous and non-radioactive wastes will be disposed of in a manner consistent with good industrial practice and in accordance with all applicable regulations. A complete estimate of the wastes and effluents to be generated during decommissioning will be provided in the applicant's plan for completion of decommissioning, to be submitted to the NRC at the time of decommissioning.

Currently there are no facilities designed and equipped for the disposition of large volumes of depleted uranium originating from enrichment facilities. The Department of Energy (DOE) currently possesses essentially the entire depleted UF_6 inventory in the United States. In July 1993, the United States Enrichment Corporation (USEC) took over from DOE low enriched uranium production activities conducted at the two operating gaseous diffusion plants (GDP) located in Portsmouth, Ohio and Paducah, Kentucky. Currently neither DOE nor USEC has in place a plan concerning final disposition of the DUF_6. The Energy Policy Act of 1992 requires DOE to address this issue. The NRC staff believes that it is premature to require a prescriptive resolution prior to DOE's determination on disposition of DUF_6, which will, to a large extent, determine the disposition options for LES' DUF_6. For the purpose of estimating funding requirements related to the disposition of DUF_6, the NRC staff finds acceptable the applicant's estimates based on conversion of DUF_6 to U_3O_8, which is much more environmentally stable than UF_6 or uranium tetrafluoride (UF_4), and disposition in a deeper than shallow land burial facility (for example, an abandoned mine cavity).

The applicant states in Section 11.8.1.2.7 of the SAR that Confidential and Restricted Data components and documents on site shall be disposed of in accordance with the requirements of 10 CFR Part 95 (NRC, 1980b). Classified portions of the centrifuges will be destroyed, piping will likely be smelted, documents will be destroyed, and other items will be handled in an appropriate manner. By license condition, a certification of nonpossession of Confidential and Restricted Data components and documents on site will be submitted to the NRC Division of Security. Details will be provided in the applicant's CEC Security Plan for the Protection of Classified Matter and Information, submitted separately in accordance with 10 CFR Part 95 and not releasable to the public.

The NRC staff accepts the applicant's commitment by license condition on the disposition of Confidential and Restricted Data components and documents.

Final Radiation Survey

The applicant states in Section 11.8.1.2.8 of the SAR that it will perform a final radiation survey to verify proper decontamination to allow the site to be released for unrestricted use. The preoperational environmental monitoring program will provide data on the natural background radiation of the area which can be used to determine any increase in levels of radiation.

The applicant states in Section 11.8.1.2.8 of the SAR that radioactivity over the entire site will systematically be measured in the final survey. The intensity of the survey will vary depending on the location (that is, the buildings, the immediate area around the buildings, the controlled fenced area, and the remainder of the site). The survey procedures and results will be documented in a report. The report will include, among other things, a map of the survey site, measurement results, and the site's relationship to the surrounding area. If the results are above allowable residual radioactivity limits, further decontamination will be performed until the results are determined to be below limits.

The NRC staff accepts the applicant's commitment to perform a final survey. NRC staff will review the results of the final survey and may perform a confirmatory survey to ensure that the facility and site can be released for unrestricted use.

15.2.1.3 Management/Organization

The applicant states in Section 11.8.1.3 of the SAR that management of the decommissioning program will ensure that proper training and procedures are provided to protect worker health and safety. The programs will focus heavily on minimizing waste volumes and worker exposure to hazardous and radioactive materials. Contractors assisting with decommissioning will likewise be subject to CEC training requirements and procedural controls.

The NRC staff accepts the applicant's general plans regarding:

- responsibilities of management of the decommissioning program
- minimization of waste volumes and worker exposures
- procedural control and training requirements for contractors assisting in decommissioning.

Details related to these three items are typically provided in the detailed decommissioning plan at the time of decommissioning.

15.2.1.4 Health and Safety

The applicant states in Section 11.8.1.4 of the SAR that as with normal operation, during decommissioning the policy shall be to keep individual and collective occupational radiation exposures as low as is reasonably achievable (ALARA). A health physics program will identify and control sources of radiation, establish worker protection requirements, and direct the use of survey and monitoring instruments.

The NRC staff accepts the applicant's statement on the responsibility of the health physics program to identify and control sources of radiation, establish worker protection requirements, and direct use of radiation survey and monitoring instruments.

15.2.1.5 Waste Management

The applicant states in Section 11.8.1.5 of the SAR that radioactive and hazardous wastes produced during decommissioning will be collected, handled, and disposed of in accordance with all regulations applicable to the CEC at the time of decommissioning. Generally, procedures will be similar to those described for wastes produced during normal operation. These wastes will ultimately be disposed of in licensed radioactive or hazardous waste disposal facilities located elsewhere. Non-hazardous and non-radioactive wastes will be disposed of consistent with good industrial practice, and in accordance with applicable regulations.

The NRC staff accepts the applicant's general commitment to collect, handle and dispose waste produced during decommissioning in accordance with applicable regulations.

15.2.1.6 Security/Material Control

The applicant states in Section 11.8.1.6 of the SAR that requirements for physical security and for material control and accounting (MC&A) will be maintained, as required during decommissioning, in a manner similar to the programs in force during operation. The plan for completion of decommissioning, submitted near the end of plant life, will provide a description of any necessary revisions to these programs.

The NRC staff accepts the applicant's proposal to maintaining requirements during decommissioning for physical security and for MC&A in a manner similar to the programs in force during operation, and that the detailed decommissioning plan submitted at the time of decommissioning will include any necessary revisions to these programs.

15.2.1.7 Recordkeeping

The applicant states in Section 11.8.1.7 of the SAR that, records important for safe and effective decommissioning of the facility will be kept in the applicant's files. Information maintained in these records will include:

- Records of spills or other unusual occurrences (including leaks of radioactive liquids) involving the spread of contamination in and around the facility, equipment, or site

- As-built drawings and modifications of structures and equipment in areas where radioactive materials are used and/or stored, including locations which possibly could be inaccessible

- Records of the cost estimate performed for the decommissioning funding plan, and records of the funding method used for assuring funds.

The NRC staff accepts the applicant's proposal to maintaining records related to the topics mentioned above. In addition, the applicant is required by license condition to also maintain records of plant and environmental radiation surveys.

15.2.1.8 Decontamination

The following paragraphs discuss the facilities, procedures, and expected results of decontamination as described by the applicant in Section 11.8.2 of the SAR. SER Table 15.1 lists all major components expected to need decontamination onsite.

The applicant states in Section 11.8.2 of the SAR that the primary contamination throughout the plant will be in the form of small amounts of UO_2F_2, with even smaller amounts of UF_4 and other compounds. The NRC staff agrees with the applicant that at the time of decommissioning, UO_2F_2 is expected to be the primary radioactive contaminant throughout the plant. Radiological contamination will be characterized by the applicant prior to commencing decommissioning activities. NRC staff review of the final decommissioning plan will include a review of the nature and extent of contamination present at the time of decommissioning.

The applicant states in Section 11.8.2.2 of the SAR that at the end of plant life, some of the equipment, most of the buildings, and all of the outdoor areas should already be acceptable for release for unrestricted use. The applicant will confirm this by performing appropriate measurements as part of the final radiation survey. The NRC will not authorize unrestricted release of equipment, buildings, and outdoor areas unless all release criteria in effect at the time of decommissioning have been met. According to the applicant, areas incidentally contaminated during normal operation will be cleaned when the contamination is discovered. This will limit the scope of necessary decontamination at the time of decommissioning.

Facilities

The applicant states in Section 11.8.2.1 of the SAR that two decontamination facilities will be required to accommodate decommissioning. A specialized facility is needed for optimal handling of the thousands of centrifuges to be decontaminated, along with the UF_6 vacuum pumps and valves. Additionally, a general purpose facility is needed for handling the remainder of the various plant components. The applicant will most likely install these facilities in existing plant buildings (such as the Centrifuge Assembly Building).

The specialized facility is described by the applicant as having four functional areas: a disassembly area, a buffer stock area, a decontamination area, and a scrap storage area for cleaned stock. The general purpose facility may share the specialized facility decontamination area. However, due to various sizes and shapes of other plant components needing handling, the disassembly area, buffer stock areas and scrap storage areas may not be shared.

Equipment in the decontamination facilities is assumed by the applicant to include:

- Transport and manipulation equipment (2-3)
- Dismantling tables, for centrifuge externals (1)
- Sawing machines (4)
- Dismantling boxes and tanks, for centrifuge internals (1)
- Degreasers (2)
- Citric acid and demineralized water baths (6)
- Contamination monitors
- Wet blast cabinets (1)
- Crusher, for centrifuge rotors (1)
- Smelting and/or shredding equipment (1)
- Scrubbing facility

The applicant states in Section 11.8.2.1 of the SAR that the decontamination facilities provided in the Technical Services Area for normal operational needs would also be available for cleaning small items during decommissioning.

The NRC staff considers the applicant's proposed plans for these decontamination facilities adequate at this time. However, the NRC staff anticipates that following plant shutdown, NRC approval of a final decommissioning plan will be required to authorize most D&D activities. The approval will be granted by the NRC if the proposed activities are determined to be in accordance with regulations, do not result in significant risks to the workers and the public, and provide safeguards for Special Nuclear Material and Confidential and Restricted Data components and documents.

Procedures

The applicant states in Section 11.8.2.2 of the SAR that procedures for decontamination will be developed and approved by plant management to minimize worker exposure and waste volumes, and to assure work is carried out in a safe manner. If, as expected, European gas centrifuge enrichment facilities are decommissioned prior to the CEC, then the experience gained will be incorporated into the procedures to be developed by the applicant.

The NRC will assess the procedures used and the results of the final survey. A confirmatory survey is expected to be part of NRC's assessment of the final survey. The facility and site will not be released for unrestricted use unless it is demonstrated that the residual contamination is within the limits and criteria in place at the time of decommissioning.

The applicant states in Section 11.8.2.2 of the SAR that contaminated plant components will be cut up or dismantled, then processed through the decontamination facilities. Contamination of site structures will be mostly confined to specific Radiation Control Zones in the Separations Building, and will be maintained at low levels throughout plant operation by regular survey and cleaning. Permanent RCZs include the Contaminated Workroom and the Pump Disassembly Room. The applicant concludes in Section 11.8.2.2 of the SAR that due to applied washable epoxy coatings and good housekeeping practices, final decontamination of these areas is unlikely to require significant removal of surface concrete or other structural material.

At the time of decommissioning, the NRC staff will review and evaluate the proposed decontamination activities in the final decommissioning plan. NRC staff will also review the facility and site characterization data. NRC staff will authorize decontamination work if it concludes that the facility and site decontamination will not violate any safety and safeguards regulations.

The applicant states in Section 11.8.2.2 of the SAR that the centrifuges will be processed through the specialized facility. The following operations will be performed:

- Removal of external fittings

- Removal of bottom flange, motor and bearings, and collection of contaminated oil

- Removal of top flange, and withdrawal and disassembly of internals

- Degreasing of items as required

* Decontamination of all recoverable items for smelting

* Destruction of other classified portions by shredding, crushing, smelting, etc.

Results

The applicant states in Section 11.8.2.3 of the SAR that as Urenco plant experience in Europe has demonstrated, conventional decontamination techniques are effective for all plant items. Recoverable items will be decontaminated and suitable for reuse except for a very small amount of intractably contaminated material. Material requiring disposal will primarily be centrifuge rotor fragments, trash, and residue from the effluent treatment systems. The applicant does not anticipate problems which will prevent the site from being released for unrestricted use.

The NRC staff agrees with the applicant that significant problems, which might prevent the site from being released for unrestricted use, are not anticipated.

15.2.2 Financial Assurance for Decommissioning

This section discusses the applicant's estimation of decommissioning costs, and explains the arrangements made by the applicant to assure funding is available to cover these costs.

15.2.2.1 Decommissioning and Tails Disposition Costs

Table 15.2 provides a summary listing of the costs estimated by the applicant, of the major decommissioning activities described above. All costs are in 1996 dollars. As shown in the table, the estimated total cost is $518.34 million. This cost estimate does not incorporate any salvage value that may be realized with the sale of potential assets. Costs are anticipated to change between the time of license application and decommissioning. By license condition, the cost estimate will be adjusted periodically by the applicant and be available, along with its basis, to the NRC for review at least every 5 years.

The applicant's evaluation of decommissioning costs included an evaluation of current experience by one of the general partners in the project, Urenco, Ltd., at similar facilities in Europe. The applicant asserts that appropriate adjustments were made to account for cost differences associated with the performance of specific activities in the United States and to escalate to 1996 dollars.

Costs estimated by the applicant include:

Facility and Site Characterization $ 0.22 million

This is based upon Urenco's on-going experience in the decommissioning of gaseous centrifuge enrichment facilities.

Table 15.2 Estimated decommissioning and tails disposition costs and duration

Activity	Cost (Millions, 1996 $s)	Time (Yrs)
Characterize CEC Facility/Site	$ 0.22	0.50
NRC Staff Review of Facility/Site Characterization	0.05	0.33
Develop and Submit to NRC Detailed Decommissioning Plan	0.22	0.50 (c)
NRC Staff Review and Approval of Decommissioning Plan	0.05	0.33
Idle Time Between Cessation of Operations and Start of Decommissioning Activities	1.0	0.50
Decontamination Facility Installation, System Cleaning, Dismantling, Decontamination	23.10	4.00
Decontamination/Decommissioning of Decontamination Facility	1.90	(a)
Sale/Salvage	0.00	(a)
Radioactive Waste Disposal	1.40	(a)
Hazardous/Mixed Waste Disposal	0.10	(a)
Tails Disposition	485.3 (b)	(a)
LES Final Radiation Survey and NRC Confirmatory Survey	1.50	1.25
Contingency	3.50	N/A
TOTALS	$ 518.34	7.41

(a) To be performed along with dismantling and decontamination.
(b) Tails disposal costs are estimated to be $16.175 million per year of tails production.
(c) Four months overlap with NRC review of characterization.

NRC Review of Facility and
Site Characterization $ 0.05 million

This is based upon the costs incurred by the NRC to review the LES license application.

Decommissioning Plan Development $ 0.22 million

This is based upon LES' experience in developing and submitting NRC-required information.

NRC Review and Approval of
Decommissioning Plan $ 0.05 million

This is based upon the costs incurred by the NRC to review the LES license application.

Idle Time Before Decommissioning $ 1.0 million

This is based on a 6-month delay between cessation of operations and start of decommissioning activities.

Decontamination Facility Installation
System Cleaning
Dismantling
Decontamination $ 23.1 million

This is based upon over ten year's of Urenco experience in decommissioning two pilot uranium enrichment centrifuge facilities at the Almelo enrichment facility in the Netherlands.

Decontamination/Decommissioning of
Decontamination Facility $ 1.9 million

This is based upon the size of the decontamination facility.

Radioactive Waste Disposal $ 1.4 million

This assumes 100 m^3 @ $12,360 per m^3 ($350/ft^3), in 1992 dollars, escalated to 1996 dollars. This cost of disposal is estimated by the applicant specifically for radioactive waste disposal in the Central States Compact (DES, 1992a,b).

Hazardous/Mixed Waste Disposal $ 0.1 million

The applicant states that D&D processes, as described in Section 11.8 of the SAR, will not result in the production of hazardous or mixed wastes for disposal. Normal accumulation of hazardous and mixed wastes will occur during the final months of CEC operation. The volume of these final wastes, not due to D&D activities, are estimated by the applicant to be approximately equivalent to the annual amounts listed in the CEC SAR, Table 7.1-1.

Tails Disposition $ 485.3 million

Rulemaking related to the disposition of depleted uranium from enrichment facilities is likely to be developed and implemented after the CEC is licensed to operate. Once developed, it will form the basis of the applicant's financial assurance. Until that happens, the applicant's financial assurance will be based on conversion of depleted UF_6 to U_3O_8 and disposition in a deeper than shallow land burial facility (for example, an abandoned mine cavity). The applicant has committed decommissioning funding to be posted as the CEC operates (LES, 1993j). An appropriate portion of the income from the sale of enrichment services will go into an external fund which is separate from the maintenance and operations fund. The cost of conversion and final disposition of depleted uranium will be reviewed and revised, if necessary, by the applicant, every five years after being licensed by the NRC.

The annual tails disposition cost escalated to 1996 dollars is estimated by the applicant to be $16.175 million. This is multiplied by 30 years to arrive at the $485.3 million figure. Costs are based on converting UF_6 to UF_4 with subsequent conversion to U_3O_8 and deep burial at an appropriate disposition site. Conversion costs are based on estimates given to the applicant by a vendor which could make this service available to the applicant. Disposition costs of U_3O_8 are based on applicant and NRC estimates (NRC, 1993b) (MMES, 1990).

Final Radiation Survey $ 1.5 million

This figure is estimated by the applicant by two methods, as follows:

1) The first method is by extrapolation from NUREG/CR-2241, "Technology and Cost of Termination Surveys Associated With Decommissioning of Nuclear Facilities," (Witherspoon, 1982). The 1980 costs of decommissioning a fuel fabrication facility and a UF_6 production facility were escalated at 5 percent per year to 1990. The higher of the two costs, (calculated for a 1 millirem and a 5 millirem annual dose to the public), were selected and then averaged, for a total of $750,000. Further escalation brings the cost to $950,000.

2) The second estimate was roughly approximated at $725,000 in 1990 dollars, and is escalated to 1996 dollars. The estimate was based on experience, using the following assumptions:

12,000 hours for grid of property and gamma count
$23,000 for soil sampling
150 core holes for depth profile
Building size of 230m x 115 m (750 ft x 380 ft)
Workhour rate, including per diem, $60/hour
Extensive use of swipes
Final analyses and report included

Based upon the costs incurred to date to review the LES license application, the cost of the NRC's confirmatory survey of the facility and site following the final radiation survey is $0.5 million.

Contingency	$ 3.5 million

A contingency of $3.5 million has been added to the estimate to account for unanticipated happenstance.

Total Applicant Estimate	$ 518.34 million

The NRC staff has reviewed this figure and its bases. The NRC staff believes that the figure presented above is a reasonable estimate for the purpose of estimating decommissioning funding requirements. By license condition, this figure along with its bases will be reviewed and appropriately revised by the applicant, if necessary, and will be made available for NRC review every 5 years.

15.2.2.2 Funding Arrangements

The applicant intends to utilize an external trust coupled with a surety bond to provide financial assurance for decommissioning pursuant to 10 CFR 40.36(e)(3) and 10 CFR 70.25(f)(3). The trust will be used to collect decommissioning funds over the life of the plant. The surety bond will provide an ultimate guarantee that decommissioning costs will be paid in the event the applicant is unable to meet its decommissioning obligations at the time of decommissioning. The trust and surety bond instruments will become effective upon receipt of licensed material on-site.

With respect to the trust, the applicant presently intends to provide for the following attributes. First, the trust fund will be external to the applicant with fund assets derived from periodic (at least annually) contributions and administered by a trustee. Second, the trust will be governed by a trust agreement which will provide, among other restrictions, for the distribution of fund assets only upon commencement of decommissioning activities and only for the purpose of decommissioning. Further, the trust may retain property with face value, and the trustee may make reasonable prudent investments, with investment income accruing to the trust.

With respect to the Surety Bond, the applicant presently anticipates providing for the following attributes. First, the bond will be issued by a Company which is listed as a qualified surety listed in the Department of Treasury, Circular 570. Second, the bond will be written for a specified term and will be renewable automatically unless the issuer serves notice at least 90 days prior to expiration of an intent not to renew. Such notice must be served upon the NRC, the trustee of the Standby Trust, and the applicant. Further, in the event the applicant is unable to provide an acceptable replacement within 30 days of such notice, the full amount of the bond will be payable automatically, prior to expiration, without proof of forfeiture.

The Surety Bond will require that any funds paid under its terms will be deposited directly into the external trust or, if necessary or appropriate, a Standby Trust, by the surety company.

The NRC staff finds this guarantee method, proposed by the applicant, acceptable for meeting the requirements of 10 CFR 40.36(e)(3) and 10 CFR 70.25(f)(3). The NRC will review for approval the actual executed documents prior to receipt of licensed material at the CEC.

15.3 Conclusion

The applicant is committed by license condition to decontaminate and decommission the CEC facility and the site at the end of its operation so that the facility and grounds can be released for unrestricted use. At this time, the NRC staff finds adequate the applicant's proposed general procedures and estimated funding required to adequately decontaminate and decommission the facility and site. The applicant has committed by license condition to reviewing, and updating as appropriate, the decommissioning funding plan, at least once every 5 years starting from the time of issuance of the license. At the time of decommissioning, a plan will be prepared by the applicant in accordance with 10 CFR 70.38 and submitted to the NRC for review and approval. The plan will describe in detail the proposed decommissioning activities and procedures. At that time, the plan will also provide a better basis for estimating decommissioning costs, which the applicant has committed to revising if required.

At the time of decommissioning, the applicant has committed to:

- removing enrichment equipment; only building shells and the site infrastructure will remain

- decontaminating all remaining facilities to acceptable levels for unrestricted use

- destroying or disposing material, components, and documents in accordance with the CEC Security Plans for the Protection of Classified Matter and Information

- submitting to the NRC Division of Security, a "Certification of Non-Possession" of classified information on the CEC site in accordance with 10 CFR 95.53

- disposing radioactive wastes generated during operation of the CEC in licensed waste disposal sites

- disposing hazardous wastes and mixed-wastes in accordance with regulatory requirements.

A facility and site radiological characterization will be performed by the applicant in accordance with 10 CFR 70.38. This effort will include:

- direct gamma radiation levels at one meter from surfaces in terms of microrads per hour or equivalent

- beta and direct gamma radiation levels at one centimeter from surfaces in terms of microrads per hour or equivalent

- removable and fixed alpha radiation surface concentrations in terms of disintegrations per minute per 100 square centimeters or equivalent

- radionuclide, gross alpha, and gross beta volume concentrations in water in terms of microcuries per milliliter or equivalent, and in solids such as soils and concrete in terms of picocuries per gram or equivalent.

The detailed decommissioning plan, which will be prepared by the applicant in accordance with 10 CFR 70.38 and submitted to the NRC for review and approval will include:

- a description of the proposed D&D activities

- a description of the methods used to protect workers and the environment during D&D

- a description of the planned final radiation survey to be conducted by the licensee to demonstrate that the facility and grounds have been decontaminated and can be released for unrestricted use

- an updated cost estimate for decommissioning, and if this estimate is higher than the funds set aside, then a description of the financial arrangements made to ensure that adequate funds will be available to cover these costs to complete decommissioning.

In accordance with 10 CFR 73.67, licensees are required to submit a physical security plan or an amended physical security plan describing how they will comply with all the appropriate requirements including schedules of implementation. The physical security plan remains in effect until the close of the period for possession of special nuclear material under the license for which the original plan was submitted.

The applicant has based the decommissioning funding estimate on the costs of dismantling and decommissioning Urenco's two centrifuge pilot plants located in Almelo, The Netherlands. One plant with 8,000 centrifuges operated from 1972 to 1981, and the other with 4,400 centrifuges from 1973 to 1976. In addition, 2,100 centrifuges from other plants were decommissioned by Urenco. According to the applicant, the total cost for decommissioning 14,500 centrifuges was approximately $7.4 million (LES, 1993g). The CEC is expected to operate about 40,000 centrifuges for 30 years. The cost of similar decommissioning activities at the CEC was estimated by the applicant simply by multiplying $7.4 million by the ratio of 40,000 centrifuges to 14,500 centrifuges. The NRC staff finds this an acceptable cost estimation method. However, decommissioning of other more comparable Urenco centrifuge facilities located in Europe is anticipated to occur before decommissioning of the CEC. These activities are likely to provide additional data on which a better cost estimate may be determined.

16 COMMON DEFENSE AND SECURITY*

The applicant must comply with the requirements of Part 95 of Title 10 of the Code of Federal Regulations (CFR) (NRC, 1980b) in order to use, process, store, reproduce, transmit or handle National Security Information (NSI) and/or Restricted Data (RD) in connection with NRC-related activities. The NRC published the "Joint NRC/DOE Classification Guide for the Louisiana Energy Services Gas Centrifuge Plant" (CG-LEP-1) (NRC, 1992e) in March 1992. This document provides classification guidance for information regarding the uranium enrichment plant to be built by Louisiana Energy Services (LES) using classified gas centrifuge technology developed by Urenco for the production of low enriched uranium.

On January 29, 1991, as part of the License Application, the applicant submitted a "Security Plan for the Protection of Classified Matter and Information" to the NRC Division of Security for review. This plan provides detailed information on LES's Claiborne Enrichment Center (CEC) proposed security procedures and controls for the protection of NSI or RD. The revised plan, approved by the NRC Division of Security in April 1992, is classified and is not releasable to the public.

In addition, LES must submit a Transportation Security Plan to the NRC Division of Security for review and approval before the transportation or movement of any classified matter. The Transportation Security Plan, also based on 10 CFR Part 95, must be prepared in accordance with the NRC "Security Plan Standard Format and Content for the Protection of Classified Matter During Transportation for NRC Licensees, Licensee Contractors, Agents, and Others," which has been provided to the applicant. The NRC notified the applicant by letter dated August 17, 1993, that the Transportation Security Plan must be submitted at least a year before commencement of transportation activities. This is needed to provide enough time for NRC to review and approve the plan, to establish required transportation and security procedures, and to process and grant security clearances to involved transport personnel.

* for Safeguards Physical Security Plan, see Chapter 14

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

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

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

17 CONCLUSIONS AND LICENSE CONDITIONS

The NRC staff has reviewed the Claiborne Enrichment Center (CEC) Safety Analysis Report (SAR) (LES, 1993a) and the applicant's Proposed License Conditions (PLC) (LES, 1993e and LES, 1994) and found, generally, that the application satisfies the requirements of 10 CFR 70.9 and that the proposed designs and conditions provide reasonable assurance that construction and operation of the CEC does not pose an undue risk to public health and safety. However, in a limited number of instances, the NRC staff finds that additional clarification is useful. This chapter summarizes the license conditions which the NRC staff recommends to supplement the applicant's PLC and presents the NRC staff's conclusions. More detailed discussions of the NRC staff's recommended license conditions for Radiation Protection and Design of Structures, Systems, and Components are presented in Chapters 8 and 4, respectively.

17.1 General Conditions

The NRC staff finding that construction and operation of the CEC does not pose an undue risk to public health and safety is based, in part, on the NRC staff review of the descriptions presented in the CEC SAR. To ensure that this assumption is sound, the NRC staff recommends the following license condition:

> A plant unit is described as feed station and take-off stations, necessary support systems, and at least one cascade of gas centrifuge machines all installed and tested. Production operation of the plant unit will be permitted only after the NRC staff has inspected the installation of all systems in a plant unit essential for beginning enrichment of uranium and verified that the initial construction of the plant unit is in accord with the requirements of the license.

17.2 Radiation Protection Conditions

The applicant commits to using gloveboxes designed to maintain at least 0.1 inches of water differential pressure anytime use of the glovebox is likely to result in exceeding the "Airborne Radioactivity Area" limits of 10 CFR 20.1003, and will cease using any glovebox until the required differential pressure has been restored. Because the referenced limit (0.6 percent of the annual limit on intake in one week) could result in a weekly intake of soluble uranium in excess of 10 mg, and a differential pressure of 0.1 inches water as indicated by a gauge may not provide adequate air inleakage into the glovebox during normal operations and during abnormal events, the NRC staff recommends the following license condition:

> Notwithstanding the requirements related to gloveboxes in Section 3.2.5.1 of the applicant's Proposed License Conditions, gloveboxes shall be designed to maintain a negative differential pressure of 0.25 inches of water. This differential pressure shall be maintained anytime that the glovebox is in use. If the differential pressure is lost, use of the glovebox will cease until the required differential pressure is restored.

The applicant proposes to calibrate radiation detection instruments prior to initial use and to subsequently perform annual calibration or calibration verifications. The NRC staff finds that instrument calibration, rather than verification of the initial calibration, may be needed to ensure reasonable accuracy and reliability, and that more frequent calibrations may be recommended by the manufacturer. Thus, the NRC staff recommends the following license condition:

> Notwithstanding the instrument calibration requirements in Section 3.2.4 of the applicant's Proposed License Conditions, instruments used for radiation protection purposes shall be calibrated before initial use and undergo periodic operability checks in accordance with written, established procedures. If an instrument fails an operability check or has undergone repair or any modification that could affect it's proper response, it shall be recalibrated. Instruments shall be recalibrated at least annually or according to the manufacturer's recommendations, whichever is more frequent.

The applicant proposes to calibrate air flow measurement devices prior to initial use, and to subsequently perform annual calibration or calibration verification. The NRC staff finds that annual verification of the initial calibration of air flow meters may not ensure accuracy over the entire range of measurements. Thus, the NRC staff recommends the following license condition:

> Notwithstanding the calibration requirements for air flow measurement devices in Section 3.2.4 of the applicant's Proposed License Conditions, flow rate meters or devices used to measure flow rates for air or effluent sampling shall be calibrated in accordance with procedures at least annually and after modifications or repairs to the meter, and when the meter is believed to have been damaged.

The applicant proposes action levels for environmental monitoring program air, water, and soil samples which are acceptable to the NRC staff. The NRC staff finds that the applicant's proposed action level for vegetation may be high and thus recommends the following license condition:

> Notwithstanding the action level for gross alpha activity in Table 5.2-2 of the applicant's Proposed License Conditions, the action level for gross alpha activity in vegetation collected in the environmental monitoring program shall not exceed 1.85×10^{-4} Bq/g (0.005 pCi/g).

17.3 Design of Structures, Systems, and Components Conditions

The NRC staff's finding on acceptability of the applicant's proposed sampling autoclave is based, in part, on review of a specification rather than a design. In order for NRC staff to confirm that the final design of the sampling autoclave is in accord with the specification

(SP-539000-40-3) reviewed in the SER, the NRC staff recommends the following license condition:

> As an element of the required preoperational inspection process, the applicant will supply materials described in FDI Specification SP-539000-40-3 to the NRC for review and approval. The materials, identified in Section 1.6 of the specification, shall include :
>
> A. Drawings including dimension drawings and hydraulic connection drawings
>
> B. Technical data including (1) design calculations, (2) descriptive literature, and (3) material certifications
>
> C. ASME Code documents and special requirements, including (1) ASME forms in accordance with Section VIII, Division 1, (2) hydrostatic test results, (3) photograph of nameplate, and (4) the seller's QA plan consistent with 10 CFR Part 50, Appendix B and NQA-1.
>
> Items A and B (1) must be the final delivered design and be complete and in sufficient detail to permit a second party review.

17.4 Conclusions

The NRC staff has reviewed the applicant's SAR and supporting documentation, including responses to NRC requests for additional information, and concludes that the applicant's descriptions, specifications, and analyses provide an adequate basis for safety review of facility operations. Further, the NRC staff concludes, on the basis of the NRC staff review of the applicant's submissions and independent NRC staff analyses, that construction and operation of the facility does not pose an undue risk to public health and safety. In order to provide additional assurance that the bases for this conclusion remain unchanged, the NRC staff will perform preoperational inspections.

The review followed the Advance Notice of Proposed Rulemaking for 10 CFR Part 76 (ANPR) (NRC, 1988a) framework in identifying and evaluating those elements of plant design and operation, termed important to safety, which must function at the highest level of reliability. The function of these and related systems was evaluated for response to design basis events. Particular attention is given to criticality safety, which is evaluated in its administrative, design, and operational aspects. Normal operational impacts are assessed for maximally exposed individuals and the surrounding population. The potential consequences of a set of accidents are estimated to identify the range of potential adverse impacts and to identify required limits for operation. Where the applicant's design or procedures should be supplemented the NRC staff has recommended license conditions to provide additional assurance of safe operation.

18 SAFETY EVALUATION REPORT PREPARERS

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20 ABBREVIATIONS AND ACRONYMS

ACGIH	American Council of Governmental Industrial Hygienists
ALARA	As Low As is Reasonably Achievable
Al_2O_3	Aluminum Oxide
AMAD	Activity Median Aerodynamic Diameter
ANPR	Advance Notice Of Proposed Rulemaking
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
ATC	Applied Technology Council
BNFL	British Nuclear Fuels, Limited
B&PV	Boiler and Pressure Vessel
Bq	Becquerel
Bq/g	Becquerel per Gram
Bq/m^3	Becquerel per Cubic Meter
BSSC	Building Seismic Safety Council
BTU/s	British Thermal Unit per Second
C	Centigrade
CAA	Controlled Access Area
CAB	Centrifuge Assembly Building
CAM	Continuous Air Monitor
CCR	Central Control Room
CEC	Claiborne Enrichment Center
CEDE	Committed Effective Dose Equivalent
cfm	Cubic Feet per Minute
CFR	Code of Federal Regulations
cm^2	Square Centimeters
COE	Corp Of Engineers
CPT	Cone Penetrometer Test
CRDB	Cylinder Receipt and Dispatch Building
CSER	Criticality Safety Engineering Report
D_2O	Heavy Water
D&D	Decontamination and Decommissioning
DBE	Design Basis Earthquake
DBFL	Design Basis Flood Level
DBT	Design Basis Tornado
DEIS	Draft Environmental Impact Statement
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
ER	Environmental Report
ETQS	Employee Training and Qualifications Services
F	Fahrenheit

FCN	Facility Change Notice
FDI	Fluor Daniel, Incorporated
FNMC	Fundamental Nuclear Material Control
FSRC	Facility Safety Review Committee
ft^3	Cubic Feet
ft^3/s	Cubic Feet per Second
g	Acceleration of Gravity
GET	General Employee Training
gm	Grams
gm/m^3	Grams per Cubic Meter
gm/s	Grams per Second
gm/yr	Grams per Year
GEVS	Gaseous Effluent Vent System
gpm	Gallons Per Minute
H/U	Hydrogen to Uranium Ratio
HEPA	High Efficiency Particulate Air
HF	Hydrogen Fluoride
HVAC	Heating, Ventilation, and Air Conditioning
Hz	Hertz
ICRP	International Commission on Radiological Protection
IS	Integrated Scheduling
J/s	Joule per Second
k_{eff}	Criticality Constant, Effective
kg/yr	Kilograms per Year
kips	Thousands of Pounds Force
k_{inf}	Criticality Constant, Infinite
kPa	Thousands of Pascals
ksi	Thousands of Pounds Force Per Square Inch
kV	Thousands of Volts
kW	Thousands of Watts
l/min	Liters per Minute
LCC	Local Control Center
LES	Louisiana Energy Systems
LEU	Low Enriched Uranium
LLD	Lower Limit of Detection
LP&L	Louisiana Power and Light
LWDS	Liquid Waste Disposal System
LWDSP	Louisiana Water Discharge System Permit
m/s	Meter per Second
m^3	Cubic Meter
m^3/min	Cubic Meters per Minute
m^3/s	Cubic Meters per Second
mbar	Millibar
MC&A	Material Control and Accounting

μCi	Microcurie
μCi/ml	Microcuries per Milliliter
μCi/yr	Microcuries per Year
μg/l	Micrograms per Liter
MCW	Machine Cooling Water
mg	Milligram
mg/m^3	Milligrams per Cubic Meter
MJ	Millions of Joules
ml	Milliliter
MOU	Memorandum Of Understanding
MPa	Millions of Pascals
MPCW	Main Plant Cooling Water
mph	Miles Per Hour
mrem	Millirem
MSL	Mean Sea Level
mSv/yr	Millisieverts per year
NaF	Sodium fluoride
NaOH	Sodium Hydroxide
NCRP	National Council on Radiation Protection and Measurements
NEHRP	National Earthquake Hazards Reduction Program
NERC	National American Energy Research Council
NFPA	National Fire Protection Agency
NHC	National Hurricane Center
NOAA	National Oceanic and Atmospheric Administration
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NSI	National Security Information
NWS	National Weather Service
OJT	On the Job Training
O&M	Operating and Maintenance
OSHA	Occupational Safety and Health Administration
PC	Personal Computer
pCi/cm^2	Picocuries per Square Centimeter
pCi/gm	Picocuries per Gram
pCi/l	Picocuries per Liter
Pa	Pascals
Pa-234	Protactinium-234, an Isotope of Protactinium
PLC	Proposed License Condition
PMF	Probable Maximum Flood
ppm	Parts Per Million
psf	Pounds force per Square Foot
psia	Pounds Force Per Square Inch, Absolute
psig	Pounds Force Per Square Inch, Gauge
QA	Quality Assurance

RAI	Request for Additional Information
RCA	Radiation Control Area
RCZ	Radiation Control Zone
RD	Restricted Data
RG	Regulatory Guide
RSC	Radiation Safety Committee
RVT	Random Vibration Theory
s	Second
s/m^3	Seconds per Cubic Meter
SAR	Safety Analysis Report
SCFM	Standard Cubic Feet per Minute
SCS	U.S. Soil Conservation Service
SCWS	Spray Cooling Water System
SEID	Standard Error of Inventory Difference
SER	Safety Evaluation Report
SM	Source Material
SNM	Special Nuclear Material
SPF	Standard Project Flood
SPS	Standard Project Storm
SRD	Shipper Receiver Differences
STS	Sewage Treatment System
STEL	Short Term Exposure Limit
SWU	Separative Work Unit
TEDE	Total Effective Dose Equivalent
Th-234	Thorium-234, an Isotope of Thorium
TS	Technical Support
TSA	Technical Services Area
TWA	Time Weighted Average
U	Uranium
U-234	Uranium-234, an Isotope of Uranium
U-235	Uranium-235, an Isotope of Uranium
U-238	Uranium-238, an Isotope of Uranium
UF_4	Uranium Tetrafluoride
UF_6	Uranium Hexafluoride
UO_2F_2	Uranyl Fluoride
U_3O_8	Triuranium Octoxide
UPS	Uninterruptable Power Supply
USGS	U.S. Geological Survey
V	Volt
wt %	Weight Percent
χ/Q	Atmospheric Concentration Per Unit Source
yr	Year

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11. ABSTRACT (200 words or less)

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff safety review and evaluation of the Louisiana Energy Services, L.P. (LES, the applicant) application for a license to possess and use byproducts, source, and special nuclear material and to enrich natural uranium to a maximum of five percent U-235 by the gas centrifuge process. The plant, to be known as the Claiborne Enrichment Center (CEC), would be constructed near the town of Homer in Claiborne Parish, Louisiana. At full production in a given year, the plant will receive approximately 4,700 tonnes of feed UF_6, and produce 870 tonnes of low-enriched UF_6 and 3,830 tonnes of depleted UF_6 tails. Facility construction, operation, and decommissioning are expected to last five, thirty, and seven years, respectively.

The objective of the review is to evaluate the potential adverse impacts of operation of the facility on worker and public health and safety under both normal operating and accident conditions. The review also considers the management organization, administrative programs, and financial qualifications provided to assure safe design and operation of the facility. The NRC staff concludes that the applicant's descriptions, specifications, and analyses provide an adequate basis for safety review of facility operations and that construction and operation of the facility does not pose an undue risk to public health and safety.

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