Revision 2, July 2004

NATIONAL ENRICHMENT FACILITY

SAFETY ANALYSIS REPORT

TABLE OF CONTENTS

Page

4.0 RADIATION PROTECTION....4.0-1

4.1 COMMITMENT TO RADIATION PROTECTION PROGRAM IMPLEMENTATION....4.1-1

4.1.1 Responsibilities of Key Program Personnel....4.1-2
4.1.1.1 Plant Manager....4.1-2
4.1.1.2 Health, Safety and Environment Manager....4.1-3
4.1.1.3 Radiation Protection Manager....4.1-3
4.1.1.4 Operations Manager....4.1-4
4.1.1.5 Facility Personnel....4.1-4
4.1.2 Staffing of the Radiation Protection Program....4.1-4
4.1.3 Independence of the Radiation Protection Program....4.1-4
4.1.4 Radiation Safety Committee....4.1-5

4.2 COMMITMENT TO AN ALARA PROGRAM....4.2-1

4.2.1 ALARA Committee....4.2-2

4.3 ORGANIZATION AND PERSONNEL QUALIFICATIONS....4.3-1

4.4 COMMITMENT TO WRITTEN PROCEDURES....4.4-1

4.4.1 Radiation Work Permit Procedures....4.4-1

4.5 TRAINING COMMITMENTS....4.5-1

4.5.1 Radiation Protection Training....4.5-2

4.6 VENTILATION AND RESPIRATORY PROTECTION PROGRAMS COMMITMENTS....4.6-1

4.6.1 Ventilation Program....4.6-1
4.6.2 Respiratory Protection Program....4.6-3

4.7 RADIATION SURVEYS AND MONITORING PROGRAMS COMMITMENTS....4.7-1

4.7.1 Radiological Zones....4.7-4
4.7.1.1 Unrestricted Area....4.7-4
4.7.1.2 Restricted Area....4.7-4
4.7.1.3 Controlled Area....4.7-5
4.7.2 Access and Egress Control....4.7-6
4.7.3 Posting for Radiation Protection Awareness....4.7-6
4.7.4 Protective Clothing and Equipment....4.7-6
4.7.5 Personnel Monitoring for External Exposures....4.7-7
4.7.6 Personnel Monitoring for Internal Exposures....4.7-8
4.7.7 Evaluation of Doses....4.7-8
4.7.8 Monitor Stations....4.7-9
4.7.9 Locker Rooms....4.7-9
4.7.10 Storage Areas....4.7-9

TABLE OF CONTENTS

Page

4.8 CONTAMINATION AND RADIATION CONTROL....4.8-1

4.8.1 Internal Exposures....4.8-1
4.8.1.1 Bioassay....4.8-2
4.8.1.2 Air Monitoring and Sampling....4.8-2
4.8.2 External Exposures....4.8-3
4.8.3 Procedures....4.8-4
4.8.4 Instrumentation....4.8-4
4.8.4.1 Friskers....4.8-4
4.8.4.2 Hand and Foot Monitors....4.8-5
4.8.5 Contamination Control....4.8-5
4.8.5.1 Surface Contamination....4.8-5

4.9 MAINTENANCE AREAS-METHODS AND PROCEDURES FOR CONTAMINATION CONTROL....4.9-1

4.9.1 Decontamination Workshop....4.9-1
4.9.2 Laundry System....4.9-2

4.10 DECONTAMINATION POLICY AND PROVISIONS....4.10-1

4.11 ADDITIONAL PROGRAM COMMITMENTS....4.11-1

4.11.1 Leak-Testing Byproduct Material Sources....4.11-1
4.11.2 Records and Reports....4.11-1

4.12 REFERENCES....4.12-1

LIST OF TABLES

Table 4.1-1	Administrative Radiation Exposure Limits
Table 4.1-2	Estimated Dose Rates
Table 4.1-3	Estimated Individual Exposures
Table 4.7-1	Radiation Emitted from Natural UF_6 Feed
Table 4.11-1	Typical Quantities of Byproduct Material for a Urenco Uranium Enrichment Centrifuge Plant

LIST OF FIGURES

Figure 4.7-1 Uranium and Decay Products of Interest
Figure 4.7-2 Projected Radiological Zones

4.0 RADIATION PROTECTION

This chapter describes the facility Radiation Protection Program. The Radiation Protection Program protects the radiological health and safety of workers and complies with the regulatory requirements in 10 CFR 19 (CFR, 2003a), 20 (CFR, 2003b) and 70 (CFR, 2003c).

This chapter includes radiation protection measures that are consistent with those previously submitted for Nuclear Regulatory Commission (NRC) review in Section 8 of the Louisiana Energy Services (LES) Claiborne Enrichment Center Safety Analysis Report (LES, 1993). These measures received regulatory approval in NUREG-1491, Safety Evaluation Report for the Claiborne Enrichment Center (NRC, 1994).

The information provided in this chapter, the corresponding regulatory requirement and the NRC acceptance criteria from NUREG-1520 (NRC, 2002), Chapter 4 are summarized in the table below. Information beyond that required by the Standard Review Plan is included. This additional information is an update of that previously submitted for the Claiborne Enrichment Center, as noted above.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 4 Reference
Section 4.1 Commitment to Radiation Protection Program Implementation	10 CFR 20.1101, Subpart B	4.4.1.3
Section 4.2 Commitment to an ALARA Program	10 CFR 20.1101	4.4.2.3
Section 4.3 Organization and Personnel Qualifications	10 CFR 70.22	4.4.3.3
Section 4.4 Commitment to Written Procedures	10 CFR 70.22(8)	4.4.4.3
Section 4.5 Training Commitments	10 CFR 19.12 & 10 CFR 20.2110	4.4.5.3
Section 4.6 Ventilation and Respiratory Protection Programs Commitments	10 CFR 20, Subpart H	4.4.6.3
Section 4.7 Radiation Surveys and Monitoring Programs Commitments	10 CFR 20, Subparts F, C, L, M	4.4.7.3
Section 4.8 Contamination and Radiation Control	N/A	N/A
Section 4.9 Maintenance Areas - Methods and Procedures for Contamination Control	N/A	N/A
Section 4.10 Decontamination Policy and Provisions	N/A	N/A
Section 4.11 Additional Program Commitments	N/A	4.4.8.3

4.1 COMMITMENT TO RADIATION PROTECTION PROGRAM IMPLEMENTATION

The radiation program meets the requirements of 10 CFR 20 (CFR, 2003b), Subpart B, Radiation Protection Programs, and is consistent with the guidance provided in Regulatory Guide 8.2, Guide for Administrative Practice in Radiation Monitoring (NRC, 1973a). The facility develops, documents and implements its Radiation Protection Program commensurate with the risks posed by a uranium enrichment operation. The facility uses, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA). The radiation program content and implementation are reviewed at least annually as required by 10 CFR 20.1101(c) (CFR, 2003d). In addition, in accordance with 10 CFR 20.1101(d) (CFR, 2003d) constraints on atmospheric releases are established for the NEF such that no member of the public would be expected to receive a total effective dose equivalent in excess of 0.1 mSv/yr (10 mrem/yr) from these releases.

The facility's philosophy for radiation protection is reflected in the establishment of a Radiation Protection Program that has the specific purpose of maintaining occupational radiation exposures ALARA. This program includes written procedures, periodic assessments of work practices and internal/external doses received, work plans and the personnel and equipment required to help implement the ALARA goal.

The facility's administrative personnel exposure limits have been set below the limits specified in 10 CFR 20 (CFR, 2003b). This provides assurance that legal radiation exposure limits are not exceeded and that the ALARA principle is emphasized. The facility administrative exposure limits are given in Table 4.1-1, Administrative Radiation Exposure Limits. Estimates of the facility area radiation dose rates and individual personnel exposures, during normal operations, are shown in Table 4.1-2, Estimated Dose Rates and Table 4.1-3, Estimated Individual Exposures. These estimates are based upon the operating experience of similar Urenco facilities in Europe.

The annual dose equivalent accrued by a typical radiation worker at a uranium enrichment plant is usually low. At the Urenco Capenhurst plant, the maximum annual worker dose equivalent was 3.1 mSv (310 mrem), 2.2 mSv (220 mrem), 2.8 mSv (280 mrem), 2.7 mSv (270 mrem) and 2.3 mSv (230 mrem) during the years 1998 through 2002, respectively. For each of these same years, the average annual worker dose equivalent was approximately 0.2 mSv (20 mrem) (Urenco, 2000; Urenco, 2001; Urenco, 2002).

Protection of plant personnel requires (a) surveillance of and control over the radiation exposure of personnel; and (b) maintaining the exposure of all personnel not only within permissible limits, but "as low as is reasonably achievable," in compliance with applicable regulations and license conditions. The objectives of Radiation Protection are to prevent acute radiation injuries (nonstochastic or deterministic effects) and to limit the potential risks of probabilistic (stochastic) effects (which may result from chronic occupational exposure) to an acceptable level.

The radiation exposure policy and control measures for personnel are set up in accordance with requirements of 10 CFR 20 (CFR, 2003b) and the guidance of applicable Regulatory Guides. Recommendations from the International Commission on Radiological Protection (ICRP) and

the National Council on Radiation Protection and Measurements (NCRP) may also be used in the formulation and evolution of the facility Radiation Protection Program.

The facility corrective action process is implemented if (1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; or if an incident results in airborne occupational exposures exceeding the administrative limits or (2) the dose limits in 10 CFR 20 (CFR, 2003b), Appendix B or 10 CFR 70.61 (CFR, 2003e) are exceeded.

The information developed from the corrective action process is used to improve radiation protection practices and to preclude the recurrence of similar incidents. If an incident as described in item two above occurs, the NRC is informed of the corrective action taken or planned to prevent recurrence and the schedule established by the facility to achieve full compliance. The corrective action process and incident investigation process are described in Section 11.6, Incident Investigations and Corrective Action Process.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the general guidelines of the occupational radiation protection program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.

4.1.1 Responsibilities of Key Program Personnel

In this section the Radiation Protection Program's organizational structure is described. The responsibilities of key personnel are also discussed. These personnel play an important role in the protection of workers, the environment and implementation of the ALARA program. Chapter 2, Organization and Administration, discusses the facility organization and administration in further detail. Section 2.2, Key Management Positions of Chapter 2, presents a detailed discussion of the responsibilities of key management personnel.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the responsibilities assigned to facility personnel and the extent of incorporation of the ALARA principle into the facility's radiation protection program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994) Section 8.3.

4.1.1.1 Plant Manager

The Plant Manager is responsible for all aspects of facility operation, including the protection of all persons against radiation exposure resulting from facility operations and materials, and for compliance with applicable NRC regulations and the facility license.

4.1.1.2 Health, Safety and Environment Manager

The Health, Safety, and Environment (HS&E) Manager reports to the Plant Manager and has the responsibility for directing the activities that ensure the facility maintains compliance with appropriate rules, regulations, and codes. This includes HS&E activities associated with nuclear safety, radiation protection, chemical safety, environmental protection, and industrial safety. The HS&E Manager works with the other facility managers to ensure consistent interpretations of HS&E requirements, performs independent reviews and supports facility and operations change control reviews.

4.1.1.3 Radiation Protection Manager

The Radiation Protection Manager reports to the HS&E Manager. The Radiation Protection Manager is responsible for implementing the Radiation Protection Program. In matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager. The Radiation Protection Manager and his staff are responsible for:

- Establishing the Radiation Protection Program
- Generating and maintaining procedures associated with the program
- Assuring that ALARA is practiced by all personnel
- Reviewing and auditing the efficacy of the program in complying with NRC and other governmental regulations and applicable Regulatory Guides
- Modifying the program based upon experience and facility history
- Adequately staffing the Radiation Protection group to implement the Radiation Protection Program
- Establishing and maintaining an ALARA program
- Establishing and maintaining a respirator usage program
- Monitoring worker doses, both internal and external
- Complying with the radioactive materials possession limits for the facility
- Handling of radioactive wastes when disposal is needed
- Calibration and quality assurance of all radiological instrumentation, including verification of required Lower Limits of Detection or alarm levels
- Establishing and maintaining a radiation safety training program for personnel working in Restricted Areas

- Performing audits of the Radiation Protection Program on an annual basis
- Establishing and maintaining the radiological environmental monitoring program
- Posting the Restricted Areas, and within these areas, posting: Radiation, Airborne Radioactivity, High Radiation and Contaminated Areas as appropriate; and developing occupancy guidelines for these areas as needed.

4.1.1.4 Operations Manager

The Operations Manager is 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 facility personnel meet the limits specified in applicable regulations, procedures and guidance documents.

4.1.1.5 Facility Personnel

Facility personnel are required to work safely and to follow the rules, regulations and procedures that have been established for their protection and the protection of the public. Personnel whose duties require (1) working with radioactive material, (2) entering radiation areas, (3) controlling facility operations that could affect effluent releases, or (4) directing the activities of others, are trained such that they understand and effectively carry out their responsibilities.

4.1.2 Staffing of the Radiation Protection Program

Only suitably trained radiation protection personnel are employed at the facility. For example, the Radiation Protection Manager has, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and three years of responsible nuclear experience associated with implementation of a Radiation Protection Program. At least two years of this nuclear experience is at a facility that processes uranium, including uranium in soluble form. Other members of the Radiation Protection Program staff are trained and qualified consistent with the guidance provided in American National Standards Institute (ANSI) standard 3.1, Selection, Qualification and Training of Personnel for Nuclear Power Plants (ANSI, 1993a).

Sufficient resources in terms of staffing and equipment are provided to implement an effective Radiation Protection Program.

4.1.3 Independence of the Radiation Protection Program

The Radiation Protection Program remains independent of the facility’s routine operations. This independence ensures that the Radiation Protection Program maintains its objectivity and is focused only on implementing sound radiation protection principles necessary to achieve occupational doses and doses to members of the public that are ALARA. It was previously

noted in Section 4.1.1.3, Radiation Protection Manager, that in matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager.

4.1.4 Radiation Safety Committee

A Radiation Safety Committee meets periodically to review, in accordance with 10 CFR 20.1101(c) (CFR, 2003d), the status of projects, measure performance, look for trends and to review radiation safety aspects of facility operations. The Radiation Protection Manager chairs the Radiation Safety Committee. The other Radiation Safety Committee members come from quality assurance, operations, maintenance, and technical support, as deemed appropriate by the Plant Manager.

The objectives of the Radiation Safety Committee are to maintain a high standard of radiation protection in all facility operations. The Radiation Safety Committee reviews the content and implementation of the Radiation Protection Program at a working level and strives to improve the program by reviewing exposure trends, the results of audits, regulatory inspections, worker suggestions, survey results, exposure incidents, etc.

The maximum interval between meetings may not exceed 180 days. A written report of each Radiation Safety Committee meeting is forwarded to all Managers.

4.2 COMMITMENT TO AN ALARA PROGRAM

Section 4.1, Commitment to Radiation Protection Program Implementation, above states the facility's commitment to the implementation of an ALARA program. The objective of the program is to make every reasonable effort to maintain facility exposures to radiation as far below the dose limits of 10 CFR 20.1201 (CFR, 2003f) as is practical and to maintain radiation exposures to members of the public such that they are not expected to receive the dose limits of 10 CFR 20.1101(d) (CFR, 2003d). The design and implementation of the ALARA program is consistent with the guidance provided in Regulatory Guides 8.2 (NRC, 1973a), 8.13 (NRC, 1999a), 8.29 (NRC, 1996), and 8.37 (NRC, 1993g). The operation of the facility is consistent with the guidance provided in Regulatory Guide 8.10 (NRC, 1977).

Annual doses to individual personnel are maintained ALARA. In addition, the annual collective dose to personnel (i.e., the sum of all annual individual doses, expressed in person-Sv or person-rem) is maintained ALARA. The dose equivalent to the embyro/fetus is maintained below the limits of 10 CFR 20.1208 (CFR, 2003g).

The Radiation Protection Program is written and implemented to ensure that it is comprehensive and effective. The written program documents policies that are implemented to ensure the ALARA goal is met. Facility procedures are written so that they incorporate the ALARA philosophy into the routine operations of the facility and ensure that exposures are consistent with 10 CFR 20.1101 (CFR, 2003d) limits. As discussed in Section 4.7, Radiation Surveys and Monitoring Programs Commitments, radiological zones will be established within the facility. The establishment of these zones supports the ALARA commitment in that the zones minimize the spread of contamination and reduce unnecessary exposure of personnel to radiation.

Specific goals of the ALARA program include maintaining occupational exposures as well as environmental releases as far below regulatory limits as is reasonably achievable. The ALARA concept is also incorporated into the design of the facility. The size and number of areas with higher dose rates are minimized consistent with accessibility for performing necessary services in the areas. Areas where facility personnel spend significant amounts of time are designed to maintain the lowest dose rates reasonably achievable.

The Radiation Protection Manager is responsible for implementing the ALARA program and ensuring that adequate resources are committed to make the program effective. The Radiation Protection Manager prepares an annual ALARA program evaluation report. The report reviews (1) radiological exposure and effluent release data for trends, (2) audits and inspections, (3) use, maintenance and surveillance of equipment used for exposure and effluent control, and (4) other issues, as appropriate, that may influence the effectiveness of the radiation protection/ ALARA programs. Copies of the report are submitted to the Plant Manager and the Safety Review Committee.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the responsibilities assigned to facility personnel and the extent of incorporation of the ALARA principle in facility's radiation protection program and concluded that the descriptions, specifications or analyses provided an adequate basis for

safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994) Section 8.3.

4.2.1 ALARA Committee

The Safety Review Committee (SRC) fulfills the duties of the ALARA Committee. The SRC meets at least quarterly. Additional details concerning the membership and qualifications of the SRC are provided in Chapter 2, Organization and Administration.

Programs for improving the effectiveness of equipment used for effluent and exposure control are also evaluated by the SRC. The recommendations of the committee are documented in writing. The implementation of the committee's recommendations is tracked to completion via the Corrective Action Program, which is described in Section 11.6, Incident Investigations and Correction Action Process.

As part of its duties, the SRC reviews the effectiveness of the ALARA program and determines if exposures, releases and contamination levels are in accordance with the ALARA concept. It also evaluates the results of assessments made by the radiation protection organization, reports of facility radiation levels, contamination levels, and employee exposures for identified categories of workers and types of operations. The committee is responsible for ensuring that the occupational radiation exposure dose limits of 10 CFR 20 (CFR, 2003b) are not exceeded under normal operations. The committee determines if there are any upward trends in personnel exposures, environmental releases and facility contamination levels.

The ALARA program facilitates interaction between radiation protection and operations personnel. The SRC, comprising staff members responsible for radiation protection and operations, is particularly useful in achieving this goal. The SRC periodically reviews the goals and objectives of the ALARA program. The ALARA program goals and objectives are revised to incorporate, as appropriate, new technologies or approaches and operating procedures or changes that could cost-effectively reduce potential radiation exposures.

4.3 ORGANIZATION AND PERSONNEL QUALIFICATIONS

The regulation 10 CFR 70.22 (CFR, 2003h) requires that the technical qualifications, including training and experience of facility staff be provided in the license application. This information is provided in this section.

The Radiation Protection Program staff is assigned responsibility for implementation of the Radiation Protection Program functions. Only suitably trained radiation protection personnel are employed at the facility. Staffing is consistent with the guidance provided in Regulatory Guides 8.2 (NRC, 1973a) and 8.10 (NRC, 1977).

As previously discussed, the Radiation Protection Manager has, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and three years of responsible nuclear experience associated with implementation of a Radiation Protection Program. The nuclear experience includes at least two years of experience at a facility that processes uranium, including uranium in soluble form. As stated in Section 4.1.2, Staffing of the Radiation Protection Program, other members of the Radiation Protection Program staff are trained and qualified consistent with the guidance provided in American National Standards Institute (ANSI) standard 3.1, Selection, Qualification and Training of Personnel for Nuclear Power Plants (ANSI, 1993a).

The Radiation Protection Manager reports to the HS&E Manager and has the responsibility for establishing and implementing the Radiation Protection Program. These duties include the training of personnel in use of equipment, control of radiation exposure of personnel, continuous determination and evaluation of the radiological status of the facility, and conducting the radiological environmental monitoring program. The facility organization chart establishes clear organizational relationships among the radiation protection staff and the other facility line managers. The facility operating organization is described in Chapter 2, Organization and Administration.

In all matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager. The Radiation Protection Manager is skilled in the interpretation of radiation protection data and regulations. The Radiation Protection Manager is also familiar with the operation of the facility and radiation protection concerns relevant to the facility. The Radiation Protection Manager is a resource for radiation safety management decisions.

4.4 COMMITMENT TO WRITTEN PROCEDURES

All operations at LES involving licensed materials are conducted through the use of procedures as required by 10 CFR 70.22(8) (CFR, 2003h). Radiation protection procedures are prepared, reviewed and approved to carry out activities related to the radiation protection program. Procedures are used to control radiation protection activities in order to ensure that the activities are carried out in a safe, effective and consistent manner. Radiation protection procedures are reviewed and revised as necessary, to incorporate any facility or operational changes or changes to the facility's Integrated Safety Analysis (ISA).

The radiation protection procedures are assigned to members of the radiation protection staff for development. Initial procedure drafts are reviewed by members of the facility staff, by personnel with enrichment plant operating experience, and other staff members as appropriate. The designated approver determines whether or not any additional, cross-disciplinary review is required. Changes to procedures are processed as follows. The writer documents the change as well as the reason for the change. The Radiation Protection Manager (or a designee who has the qualifications of the Radiation Protection Manager) reviews and approves procedures as well as proposed revisions to procedures. Final approval of the revised procedure is by the Plant Manager, or a designated alternate. Chapter 11, Management Measures, describes the program implemented for the control of procedures.

4.4.1 Radiation Work Permit Procedures

All work performed in Restricted Areas is performed in accordance with a Radiation Work Permit (RWP). The procedures controlling RWPs are consistent with the guidance provided in Regulatory Guide 8.10 (NRC, 1977). An RWP may also be required whenever the Radiation Protection Manager deems that one is necessary. Activities involving licensed materials not covered by operating procedures and where radioactivity levels are likely to exceed airborne radioactivity limits require the issuance of a RWP. Both routine and non-routine activities are performed under a RWP. The RWP provides a description of the work to be performed. That is, the RWP defines the authorized activities. The RWP summarizes the results of recent dose rate surveys, contamination surveys, airborne radioactivity results, etc. The RWP specifies the precautions to be taken by those performing the task. The specified precautions may include personal protective equipment to be worn while working (e.g., gloves, respirators, personnel monitoring devices), stay-times or dose limits for work in the area, record keeping requirements (e.g., time or dose spent on job) and the attendance of a radiation protection technician during the work. The RWP requires approval by the Radiation Protection Manager or designee. The designee must meet the requirements of Section 4.1.2, Staffing of the Radiation Protection Program. RWPs have a predetermined period of validity with a specified expiration or termination time.

Standing RWPs are issued for routinely performed activities, such as tours of the plant by shift personnel or the charging of cylinders. A Standing RWP would, for example, be used for the job evolution of cylinder charging; a new RWP is not issued each time a new cylinder is charged.

Listed below are requirements of the RWP procedures.

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

- Planned activities or changes to activities inside Restricted Areas or work with licensed materials are reviewed by the Radiation Protection Manager or designee for the potential to cause radiation exposures to exceed action levels or to produce radioactive contamination

- RWPs include requirements for any necessary safety controls, personnel monitoring devices, protective clothing, respiratory protective equipment, and air sampling equipment and the attendance of radiation protection technicians at the work location

- RWPs are posted at access points to Restricted Areas with copies of current RWPs posted at the work area location

- RWPs clearly define and limit the work activities to which they apply. A RWP is closed out when the applicable work activity for which it was written is completed and terminated

- RWPs are retained as a record at least for the life of the facility.

The subject matter discussed above is an improved version of the subject matter of Claiborne Enrichment Center SAR (LES, 1993). The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the RWP system and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on is in NUREG-1491 (NRC, 1994), Section 8.4.1.7.

4.5 TRAINING COMMITMENTS

The design and implementation of the radiation protection training program complies with the requirements of 10 CFR 19.12 (CFR, 2003i). Records are maintained in accordance with 10 CFR 20.2110 (CFR, 2003j).

The development and implementation of the radiation protection training program is consistent with the guidance provided in the following regulatory guidance documents:

- Regulatory Guide 8.10-Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable (NRC, 1977)
- Regulatory Guide 8.13-Instructions Concerning Prenatal Radiation Exposure (NRC, 1999a)
- Regulatory Guide 8.29-Instructions Concerning Risks From Occupational Radiation Exposure (NRC, 1996)
- ASTM C986-89-Developing Training Programs in the Nuclear Fuel Cycle (ASTM, 1989)
- ASTM E1168-95-Radiological Protection Training for Nuclear Facility Workers (ASTM, 1995).

All personnel and visitors entering Restricted Areas or Controlled Areas, as defined below, receive training that is commensurate with the radiological hazard to which they may be exposed. Alternatively, visitors will be provided with trained escorts who have received radiation protection training.

The level of radiation protection training is based on the potential radiological health risks associated with an employee's work responsibilities and incorporates the provisions of 10 CFR 19.12 (CFR, 2003i). In accordance with 10 CFR 19.12 (CFR, 2003i), any individual working at the facility who is likely to receive in a year a dose in excess of 1 mSv (100 mrem) is:

A. Kept informed of the storage, transfer, or use of radioactive material

B. Instructed in the health protection problems associated with exposure to radiation and radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed

C. Required to observe, to the extent within the worker's control, the applicable provisions of the NRC regulations and licenses for the protection of personnel from exposure to radiation and radioactive material

D. Instructed of their responsibility to report promptly to the facility management, any condition which may cause a violation of NRC regulations and licenses or unnecessary exposure to radiation and radioactive material

E. Instructed in the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation and radioactive material

F. Advised of the various notifications and reports to individuals that a worker may request in accordance with 10 CFR 19.13 (CFR, 2003k).

The radiation protection training program takes into consideration a worker's normally assigned work activities. Abnormal situations involving exposure to radiation and radioactive material, which can reasonably be expected to occur during the life of the facility, are also evaluated and factored into the training. The extent of these instructions is commensurate with the potential radiological health protection problems present in the work place.

Retraining of personnel previously trained is performed for radiological, chemical, industrial, and criticality safety at least annually. The retraining program also includes procedure changes, and updating and changes in required skills. Changes to training are implemented, when required, due to incidents potentially compromising safety or if changes are made to the facility or processes. Records of training are maintained in accordance with LES records management system. Training programs are established in accordance with Section 11.3, Training and Qualifications. The radiation protection sections of the training program are evaluated at least annually. The program content is reviewed to ensure it remains current and adequate to assure worker safety.

The specifics of the Radiation Protection Training are described in the following section.

4.5.1 Radiation Protection Training

Radiation protection training is highlighted to emphasize the high level of importance placed on the radiological safety of plant personnel and the public. In-depth radiation protection training is provided for the various types of job functions (e.g., production operator, radiation protection technician, contractor personnel) commensurate with the radiation safety responsibilities associated with each such position. Visitors to a Restricted Area are trained in the formal training program or are escorted by trained personnel while in the Restricted Area.

Personnel access procedures ensure the completion of formal nuclear safety training prior to permitting unescorted access into the Restricted Areas. Training sessions covering criticality safety, radiation protection and emergency procedures are conducted on a regular basis to accommodate new employees or those requiring retraining. Retraining is conducted when necessary to address changes in policies, procedures, requirements and the ISA.

Specific topics covered in the training program are listed in Chapter 11, Management Measures, Section 11.3.3.1.1. The training provided includes the requirements of 10 CFR 19 (CFR, 2003a).

Individuals attending these sessions must pass an initial examination covering the training contents to assure the understanding and effectiveness of the training. The effectiveness and adequacy of the training program curriculum and instructors are also evaluated by audits

performed by operational area personnel responsible for criticality safety and radiation protection.

Since contractor employees may perform diverse tasks in the Restricted Areas or Controlled Areas of the facility, formal training for these employees is designed to address the type of work they perform. In addition to applicable radiation safety topics, training contents may include RWPs, special bioassay sampling, and special precautions for welding, cutting, and grinding. Instructors certified by the Radiation Protection Manager conduct the radiation protection training programs.

The Radiation Protection Manager is responsible for establishing and maintaining the radiation protection training for all personnel, including contractor personnel who may be working at the facility. Records are maintained for each employee documenting the training date, scope of the training, identity of the trainer(s), any test results and other associated information.

Individuals requiring unescorted access to a Restricted Area receive annual retraining. Contents of the formal radiation protection training program are reviewed and updated as required at least every two years by the HS&E Manager and Radiation Protection Manager to ensure that the programs are current and adequate.

4.6 VENTILATION AND RESPIRATORY PROTECTION PROGRAMS COMMITMENTS

The regulations contained in 10 CFR 20 (CFR, 2003b), Subpart H, define the required elements of the facility respiratory protection and ventilation programs. This section describes the design and management measures taken to ensure that the installed ventilation and containment systems operate effectively. This section also describes the worker respiratory protection program. Chapter 3, Integrated Safety Analysis Summary, contains additional design and process information on important facility ventilation systems.

The design of the ventilation and respiratory protection programs is consistent with the guidance contained in the following documents:

- Regulatory Guide 8.24-Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication (NRC, 1979)
- ANSI N510-1980-Testing of Nuclear Air Cleaning Systems (ANSI,1980)
- ERDA 76-21-Nuclear Air Cleaning Handbook (ERDA,1976)
- NCRP Report No. 59-Operational Radiation Safety Program (NCRP,1978)
- Regulatory Guide 8.15-Acceptable Programs for Respiratory Protection (NRC,1999b)
- ANSI Z88.2-1992-Practices for Respiratory Protection (ANSI,1992).

4.6.1 Ventilation Program

The confinement of uranium and the attenuation of its associated radiation are a design requirement for the facility. The internal radiation exposure of workers is controlled primarily by the containment of UF_6 within process equipment. The entire UF_6 enrichment process, except for liquid sampling, is operated under a partial vacuum so that leaks are into the system and not into work areas.

Ventilation systems for the various buildings control the temperature and the humidity of the air inside the building. The ventilation systems serving normally non-contaminated areas exhaust approximately 10% of the air handled to the atmosphere. Ventilation systems serving potentially contaminated areas include design features that provide for confinement of radiological contamination. Ventilation systems for potentially contaminated areas exhaust 100% of the air handled to the environment through the exhaust stacks. All air released from potentially contaminated areas is filtered to remove radioactive particulates before it is released. The ventilation systems for potentially contaminated areas are designed to maintain the potentially contaminated areas at a slightly negative pressure relative to the uncontaminated areas. This ensures that the airflow direction is from areas of little or no contamination to areas

of higher contamination. Refer to Chapter 3, Integrated Safety Analysis Summary, for further information.

Process vents from the Separations Building Module are collected by the Separations Building Gaseous Effluent Vent System (GEVS). Some areas of the Technical Services Building (TSB) also have fume hoods that are connected to the TSB GEVS. Air released from the Centrifuge Test Facility and the Centrifuge Post Mortem Facilities is filtered by the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System prior to release. The systems operate slightly below atmospheric pressure to remove potentially hazardous vapors and particulate from confined areas of the plant. The systems contain particulate and carbon adsorption filters to remove radioactive materials from the gas stream prior to release from the plant. Continuous HF monitors are provided upstream of the filters with high level alarms to inform operators of UF_6 releases in the plant. Refer to Chapter 3, Integrated Safety Analysis Summary, for further information.

Normal operation of the facility will not result in a release of radioactive material that exceeds regulatory limits. Ventilation systems for areas that do not have the potential for contamination are not monitored for radioactivity because radioactive material is not handled or processed in these areas. No emergency ventilation systems are provided for operation when the normal ventilation systems are shut down. Refer to Chapter 3, Integrated Safety Analysis Summary, for additional design and process information on the facility ventilation systems.

Several measures are in place to ensure effective operation of the ventilation systems. Differential pressure across High Efficiency Particulate Air (HEPA) filters in potentially contaminated ventilation exhaust systems is monitored monthly or automatically monitored and alarmed. Operating procedures specify limits and set points on the differential pressure consistent with manufacturers' recommendations. Filters are changed if they fail to function properly or if the differential pressure exceeds the manufacturers' ratings.

Filter inspection, testing, maintenance and change out criteria are specified in written procedures approved by the Technical Services Manager, or a designated alternate. Change-out frequency is based on considerations of filter loading, operating experience, differential pressure data and any UF_6 releases indicated by HF alarms.

Gloveboxes are designed to maintain a negative differential pressure of about 0.623 mbar (0.25 in H_2O). This differential pressure is maintained anytime that the glovebox is in use. If the differential pressure is lost, use of the glovebox is suspended until the required differential pressure is restored.

Air flow rates at exhausted enclosures and close-capture points, when in use, are adequate to preclude escape of airborne uranium and minimize the potential for intake by workers. Air flow rates are checked monthly when in use and after modification of any hood, exhausted enclosure, close-capture point equipment or ventilation system serving these barriers.

The various programs that pertain to preventive and corrective maintenance are described in Chapter 11, Sections 11.2.2, Corrective Maintenance and 11.2.3, Preventive Maintenance respectively.

4.6.2 Respiratory Protection Program

The facility uses process and engineering controls to control the concentration of radioactive material in air. However, there may be instances when it is not practical to apply process or other engineering controls. When it is not possible to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area, other means are implemented to maintain the total effective dose equivalent ALARA. In these cases, the ALARA goal is met by an increase in monitoring and the limitation of intakes by one or more of the following means:

A. Control of access

B. Limitation of exposure times

C. Use of respiratory protection equipment

D. Other controls, as available and appropriate.

If an ALARA analysis is performed to determine whether or not respirators should be used, safety factors other than radiological factors may be considered. The impact of respirator use on workers' industrial health and safety is factored into decisions to use respirators.

If the decision is made to permit the use of respiratory protection equipment to limit the intake of radioactive material, only National Institute of Occupational Safety and Health (NIOSH) certified equipment is used. The respiratory protection program meets the requirements of 10 CFR 20 (CFR, 2003b), Subpart H (Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas).

The respiratory protection program includes the following elements:

A. Air sampling to identify the potential hazard, select proper equipment and estimate doses

B. Surveys and, when necessary, bioassays to evaluate actual intakes

C. Performance testing of respirators for operability (user seal check for face sealing devices and functional check for others) immediately prior to each use.

D. Written procedures for the following:

   1. Monitoring, including air sampling and bioassays

   2. Supervision and training of respirator users

   3. Fit testing

   4. Respirator selection

5. Breathing air quality

6. Inventory and control

7. Storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment

8. Record keeping

9. Limitations on periods of respirator use and relief from respirator use.

E. Determination by a physician that the individual user is medically fit to use respiratory protection equipment:

1. Before the initial fitting of a face sealing respirator

2. Before the first field use of non-face sealing respirators

3. Either every 12 months thereafter, or periodically at a frequency determined by a physician.

F. A respirator fit test requires a minimum fit factor of at least 10 times the Assigned Protection Factor (APF) for negative pressure devices, and a fit factor of at least 500 times the APF for any positive pressure, continuous flow, and pressure-demand devices. The fit testing is performed before the first field use of tight fitting, face-sealing respirators. Subsequent testing is performed at least annually thereafter. Fit testing must be performed with the facepiece operating in the negative pressure mode.

1. Each user is informed that they may leave the area at any time for relief from respirator use in the event of equipment malfunction, physical or psychological distress, procedural or communication failure, significant deterioration of operating conditions, or any other conditions that might require such relief.

2. In the selection and use of respirators, the facility provides for vision correction, adequate communication, low temperature work environments, and the concurrent use of other safety or radiological protection equipment. Radiological protection equipment is used in such a way as not to interfere with the proper operation of the respirator.

3. Standby rescue persons are used whenever one-piece atmosphere-supplying suits are in use. Standby rescue personnel are also used when any combination of supplied air respiratory protection device and personnel protective equipment is in use that presents difficulty for the wearer to remove the equipment. The standby personnel are equipped with respiratory protection devices or other apparatus appropriate for the potential hazards. The standby rescue personnel observe and maintain continuous communication with the workers (visual, voice, signal line, telephone, radio, or other suitable means). The rescue personnel are immediately available to assist the workers in case of a failure of the air supply or

for any other emergency. The Radiation Protection Manager specifies the number of standby rescue personnel that must be immediately available to assist all users of this type of equipment and to provide effective emergency rescue if needed.

4. Atmosphere-supplying respirators are supplied with respirable air of grade D quality or better as defined by the Compressed Gas Association in publication G-7.1, Commodity Specification for Air, (CGA, 1997) and included in the regulations of the Occupational Safety and Health Administration (29 CFR 1910.134(i)(1)(ii)(A) through (E) (CFR, 2003l)).

5. No objects, materials or substances (such as facial hair), or any conditions that interfere with the face-to-facepiece seal or valve function, and that are under the control of the respirator wearer, are allowed between the skin of the wearer's face and the sealing surface of a tight-fitting respirator facepiece.

The dose to individuals from the intake of airborne radioactive material is estimated by dividing the ambient air concentration outside the respirator by the assigned protection factor. If the actual dose is later found to be greater than that estimated initially, the corrected value is used. If the dose is later found to be less than the estimated dose, the lower corrected value may be used.

Records of the respiratory protection program (including training for respirator use and maintenance) are maintained in accordance with the facility records management program as described in Section 11.7, Records Management. Respiratory protection procedures are revised as necessary whenever changes are made to the facility, processing or equipment.

4.7 RADIATION SURVEYS AND MONITORING PROGRAMS COMMITMENTS

Radiation surveys are conducted for two purposes: (1) to ascertain radiation levels, concentrations of radioactive materials, and potential radiological hazards that could be present in the facility; and (2) to detect releases of radioactive material from facility equipment and operations. Radiation surveys will focus on those areas of the facility identified in the ISA where the occupational radiation dose limits could potentially be exceeded. Measurements of airborne radioactive material and/or bioassays are used to determine that internal occupational exposures to radiation do not exceed the dose limits specified in 10 CFR 20 (CFR, 2003b), Subpart C.

To assure compliance with the requirements of 10 CFR 20 (CFR, 2003b) Subpart F, there are written procedures for the radiation survey and monitoring programs. The radiation survey and monitoring programs assure compliance with the requirements of 10 CFR 20 (CFR, 2003b) Subpart F (Surveys and Monitoring), Subpart C (Occupational Dose Limits), Subpart L (Records) and Subpart M (Reports).

The radiation survey and monitoring programs are consistent with the guidance provided in the following references:

- Regulatory Guide 8.2-Guide for Administrative Practice in Radiation Monitoring (NRC,1973a)
- Regulatory Guide 8.4-Direct-Reading and Indirect-Reading Pocket Dosimeters (NRC,1973b)
- Regulatory Guide 8.7- Instructions for Recording and Reporting Occupational Radiation Exposure Data (NRC, 1992a)
- Regulatory Guide 8.9-Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (NRC,1993f)
- Regulatory Guide 8.24-Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication (NRC,1979)
- Regulatory Guide 8.25-Air Sampling in the Workplace (NRC, 1992b)
- Regulatory Guide 8.34-Monitoring Criteria and Methods To Calculate Occupational Radiation Doses (NRC, 1992c)
- NUREG-1400-Air Sampling in the Workplace (NRC,1993a)
- ANSI N13.1-1969 (R1993)-Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities (ANSI, 1993b)
- ANSI N323-1978-Radiation Protection Instrumentation Test and Calibration (ANSI,1978)

- ANSI N13.11-1983-Dosimetry-Personnel Dosimetry Performance-Criteria for Testing (ANSI, 1983)

- ANSI N13.15-1985-Radiation Detectors-Personnel Thermoluminescence Dosimetry Systems-Performance (ANSI,1985)

- ANSI/HPS N13.22-1995-Bioassay Program for Uranium (ANSI,1995)

- ANSI N13.27-1981-Performance Requirements for Pocket-Sized Alarm Dosimeters and Alarm Ratemeters (ANSI,1981)

- ANSI/HPS N13.30-1996-Performance Criteria for Radiobioassay (ANSI,1996)

- ANSI N13.6-1966 (R1989), Practice for Occupational Radiation Exposure Records Systems (ANSI,1989)

The procedures include an outline of the program objectives, sampling procedures and data analysis methods. Equipment selection is based on the type of radiation being monitored. Procedures are prepared for each of the instruments used and specify the frequency and method of calibration. Maintenance and calibration are in accordance with the manufacturers' recommendations. Specific types of instruments used in the facility are discussed below.

The survey program procedures also specify the frequency of measurements and record keeping and reporting requirements. As stated in Section 4.1, Commitment to Radiation Protection Program Implementation, the facility corrective action process is implemented if: 1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; or if an incident results in airborne occupational exposures exceeding the administrative limits, or 2) the dose limits in 10 CFR 20, Appendix B (CFR, 2003m) or 10 CFR 70.61 (CFR, 2003e) are exceeded. In the event the occupational dose limits given in 10 CFR 20 (CFR, 2003b), Subpart C are exceeded, notification of the NRC is in accordance with the requirements of 10 CFR 20, Subpart M—Reports.

All personnel who enter Restricted Areas (as defined below) are required to wear personnel monitoring devices that are supplied by a vendor that holds dosimetry accreditation from the National Voluntary Laboratory Accreditation Program. In addition, personnel are required to monitor themselves prior to exiting Restricted Areas which may have the potential for contamination.

Continuous airborne radioactivity monitors provide indication of the airborne activity levels in the Restricted Areas of the facility. Monitoring instruments for airborne alpha emitters are provided at different locations throughout facility. These monitors are designed to detect alpha emitters in the air, which would indicate the potential for uranium contamination. When deemed necessary, portable air samplers may be used to collect a sample on filter paper for subsequent analysis in the laboratory.

Monitor data is collected for regular analysis and documentation. Monitors in locations classified as Airborne Radioactivity Areas are equipped with alarms. The alarm is activated when airborne radioactivity levels exceed predetermined limits. The limits are set with

consideration being given to both toxicity and radioactivity. The volume of air sampled may have to be adjusted to ensure adequate sensitivity with minimum sampling time. The operating history of the facility, changes in technology, changes in room functions and design, and changes in regulations may necessitate adjustment of the monitors.

Continuous monitoring of direct radiation exposure rates is not performed because the uranium processed in the facility is handled in closed containers. The radionuclides of interest are primarily alpha and beta emitters. The decay data and decay chains for these radionuclides are shown in Table 4.7-1, Radiation Emitted from Natural UF_6 Feed, and Figure 4.7-1, Uranium and Decay Products of Interest, respectively.

Alpha and beta radiation cannot penetrate the container walls. Typical area radiation monitors measure gamma radiation. At this facility, the gamma radiation is not present at sufficient levels to provide representative indications. Instead, periodic radiation monitoring is performed with portable survey meters and "wipe tests" for contamination are taken to evaluate radiological conditions in the facility.

A calibration is performed in accordance with written established procedures and documented prior to the initial use of each airflow measurement instrument (used to measure flow rates for air or effluent sampling) and each radioactivity measurement instrument. Periodic operability checks are performed in accordance with written established procedures. Calibrations are performed and documented on each airflow measurement and radioactivity measurement instrument at least annually (or according to manufacturers' recommendations, whichever is more frequent) or after failing an operability check, or after modifications or repairs to the instrument that could affect its proper response, or when it is believed that the instrument has been damaged.

Unreliable instruments are removed from service until repairs are completed. Portal monitors, hand and foot monitors and friskers have the required sensitivity to detect alpha contamination on personnel to ensure that radioactive materials do not spread to the areas outside the Restricted Areas. Instruments are calibrated with sources that are within ±5% of the reference value and are traceable to the National Institute of Standards and Technology or equivalent.

The background and efficiency of laboratory counting instruments, when used for radiation protection purposes, is determined daily. This determination may be less frequent only if necessary due to long counting intervals.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the instrument, calibration and maintenance program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.6.

4.7.1 Radiological Zones

Radiological zones within the facility have been established to (1) control the spread of contamination, (2) control personnel access to avoid unnecessary exposure of personnel to radiation, and (3) control access to radioactive sources present in the facility. Table 4.1-2, Estimated Dose Rates, lists general dose rate estimates for the facility. These dose estimates were prepared based upon historical data from operating Urenco centrifuge enrichment facilities. Areas associated with higher dose rates may be restricted from public access, as determined by facility management. Areas where facility personnel spend substantial amounts of time are designed to minimize the exposure received when routine tasks are performed, in accordance with the ALARA principle.

The following definitions of areas are provided to describe how the facility Radiation Protection Program is implemented to protect workers and the general public on the site.

4.7.1.1 Unrestricted Area

NRC regulation 10 CFR 20.1003 (CFR, 2003n) defines an Unrestricted Area as an area, access to which is neither limited nor controlled by the licensee. The area adjacent to the facility site where LES does not normally exercise access control is an Unrestricted Area. This area can be accessed by members of the public, indigenous wildlife, or by facility personnel. The Unrestricted Area is governed by the limits in 10 CFR 20.1301 (CFR, 2003o). The total effective dose equivalent to individual members of the public from the licensed operation may not exceed 1 mSv (100 mrem) in a year (exclusive of background radiation). The dose in any Unrestricted Area from external sources may not exceed 0.02 mSv (2 mrem) in any one hour. In addition to the NRC limit, the Environmental Protection Agency, in 40 CFR 190 (CFR, 2003p), imposes annual dose equivalent limits of 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid, and 0.25 mSv (25 mrem) to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials to the general environment from uranium fuel cycle operations and to radiation from these operations.

4.7.1.2 Restricted Area

The NRC defines a Restricted Area as an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Access to and egress from a Restricted Area at the plant site is through a radiation protection control point known as a Monitor Station. Monitoring equipment is located at these egress points. All personnel are required to monitor themselves prior to exiting Restricted Areas that have the potential for contamination, using monitoring instruments that detect gross alpha contamination.

Examples of Restricted Areas include storage areas for UF_6 in the Cylinder Receipt and Dispatch Building and the potentially contaminated areas in the Technical Services Building. Personnel who have not been trained in radiation protection procedures are not allowed to access a Restricted Area without escort by trained personnel.

The areas defined below may exist within a Restricted Area. These areas may be temporary or permanent. The areas are posted to inform workers of the potential hazard in the area and to help prevent the spread of contamination. These areas are conspicuously posted in accordance with the requirements of 10 CFR 20.1902 (CFR, 2003q).

- An area in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.05 mSv (5 mrem) in 1 hr at 30 cm (11.8 in) from the radiation source or from any surface that the radiation penetrates is designated a "Radiation Area" as defined in 10 CFR 20.1003 (CFR, 2003n).

- An "Airborne Radioactivity Area" means a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations (1) In excess of the derived air concentrations (DACs) specified in Appendix B (CFR, 2003m), to 10 CFR 20.1001 - 20.2401, or (2) To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6% of the annual limit on intake (ALI) or 12 DAC-hours. Note that entry into this area does not automatically require the wearing of a respirator.

- A "High Radiation Area" is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1 mSv (100 mrem) in 1 hour at 30 cm (11.8 in) from the radiation source or from any surface that the radiation penetrates. No examples of this type of area are expected during routine operation of the facility. This designation is provided here only for the purposes of emergency situations (drills and actual events).

- LES defines a "Contaminated Area" as an area where removable contamination levels are above 0.33 Bq/100 cm^2 (20 dpm/100 cm^2) of alpha activity or 16.7 Bq/100 cm^2 (1,000 dpm/100 cm^2) beta/gamma activity.

The NRC limits the soluble uranium intake of an individual to 10 milligrams in a week in consideration of chemical toxicity. LES posts areas where the intake of soluble uranium in one week is likely to exceed 1 milligram, if respiratory protection is not utilized.

4.7.1.3 Controlled Area

The NRC defines a Controlled Area as an area, outside of a Restricted Area but inside the site boundary, access to which can be limited by the licensee for any reason. The area of the plant within the perimeter fence but outside any Restricted Area is part of the Controlled Area. Due to the presence of the fence, members of the public do not have direct access to this Controlled Area of the site and must be processed by security and authorized to enter the site. Training for access to a Controlled Area is provided commensurate with the radiological hazard.

Site visitors include delivery people, tour guests and service personnel who are temporary, transient occupants of the Controlled Area. Area monitoring demonstrates compliance with public exposure limits for such visitors. All individuals who are contractor or LES employees

and who work only in the Controlled Area are subject to the exposure limits for members of the public (CFR, 2003b).

4.7.2 Access and Egress Control

The facility establishes and implements an access control program that ensures that (a) signs, labels, and other access controls are properly posted and operative, (b) restricted areas are established to prevent the spread of contamination and are identified with appropriate signs, and (c) step-off pads, change facilities, protective clothing facilities, and personnel monitoring instruments are provided in sufficient quantities and locations.

Because there are no High Radiation Areas in the facility, there are no areas where access is physically prevented due to radiation level. Access control is by administrative methods. Access to certain areas may be physically prevented for security reasons. Personnel who have not been trained in radiation protection procedures are not allowed access to a Restricted Area without escort by other trained personnel.

Access to and egress from a Restricted Area is through one of the monitor stations at the particular Restricted Area boundary. Access to and egress from each Radiation Area, High Radiation Area, Contaminated Area or Airborne Radioactivity Area within a Restricted Area may also be individually controlled. A monitor (frisker), step-off pad and container for any discarded protective clothing may be provided at the egress point from certain of these areas to prevent the spread of contamination.

Action levels for skin and personal clothing contamination at the point of egress from Restricted Areas and any additional designated areas within the Restricted Area (e.g., a Contaminated Area which is provided with a step-off pad and frisker) shall not exceed 2.5 Bq/100 cm^2 (150 dpm/100 cm^2) alpha or beta/gamma contamination (corrected for background). Clothing contaminated above egress limits shall not be released unless it can be laundered to within these limits. If skin or other parts of the body are contaminated above egress limits, reasonable steps that exclude abrasion or other damage shall be undertaken to effect decontamination.

4.7.3 Posting for Radiation Protection Awareness

Restricted Areas and other areas within the Restricted Areas (e.g., Airborne Radioactivity Area) are clearly identified by physical means such as placarding or boundary marking, so that facility personnel can identify these areas and use their training to minimize their exposure. This identification is done in accordance with 10 CFR 20.1902 (CFR, 2003q). The radiation and contamination levels from the most recent survey are clearly noted on each posting.

4.7.4 Protective Clothing and Equipment

The proper use of protective clothing and equipment can minimize internal and external exposures to radioactivity. Personnel working in areas that are classified as Airborne Radioactivity Areas or Contaminated Areas must wear appropriate protective clothing. If the

areas containing the surface contamination can be isolated from adjacent work areas via a barrier such that dispersible material is not likely to be transferred beyond the area of contamination, personnel working in the adjacent area are not required to wear protective clothing. Areas requiring protective clothing are posted at each of their entry points.

Radiation protection management and associated technical staff are responsible for determining the need for protective clothing in each work area. Areas requiring protective clothing are identified by posting signs at all area entry points.

4.7.5 Personnel Monitoring for External Exposures

External exposures are received primarily from the radioactive decay products of ^{235}U and ^{238}U. Most notably these progeny are ^{231}Th (several gammas, all low energy and low abundance), ^{234}Th (several gammas, most low abundance and low energy), and ^{234}Pa and ^{234m}Pa (many gammas, variable abundance, low and high energy). The ^{234m}Pa is the primary gamma source and is expected to contribute to a significant portion of the external exposure. Over the life of the facility, the number of tails-containing Uranium Byproduct Cylinders (UBCs) placed on the storage pad may increase to the pad's design capacity. In addition, the CRDB may reach its design capacity of feed and product cylinders. As a result, it is possible that the neutron contribution to the total worker dose may require monitoring. The neutrons are due to spontaneous fission in uranium as well as the alpha, neutron reaction on fluorine. Workers receive training regarding ALARA concepts such as time-distance-shielding to minimize their exposures.

All personnel whose duties require them to enter Restricted Areas wear individual external dosimetry devices, e.g., thermoluminescent dosimeters (TLDs) that are sensitive to beta, gamma and neutron radiation. Appropriate neutron survey meters are also available to the Radiation Protection staff. External dosimetry devices are evaluated at least quarterly to ascertain external exposures. Administrative limits on radiation exposure are provided in Table 4.1-1, Administrative Radiation Exposure Limits.

If 25% of the annual administrative limit (i.e., 2.5 mSv or 250 mrem) is exceeded in any quarter, then an investigation is performed and documented to determine what types of activities may have contributed to the worker's external exposure. The administrative limit already reflects ALARA principles, so this action level is appropriate. This investigation may include, but is not limited to procedural reviews, efficiency studies of the air handling system, cylinder storage protocol, and work practices.

Anytime an administrative limit is exceeded, the Radiation Protection Manager is informed. The Radiation Protection Manager is responsible for determining the need for and recommending investigations or corrective actions to the responsible Manager(s). Copies of the Radiation Protection Manager's recommendations are provided to the Safety Review Committee.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to administrative radiation exposure limits and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the

facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.1.

4.7.6 Personnel Monitoring for Internal Exposures

Internal exposures for all personnel wearing external dosimetry devices are evaluated via direct bioassay (e.g. *in vivo* body counting), indirect bioassay (e.g., urinalysis), or an equivalent technique. For soluble (Class D) uranium, 10 CFR 20.1201(e) (CFR, 2003f) limits worker intake to no more than 10 milligrams of soluble uranium in a week. This is to protect workers from the toxic chemical effects of inhaling Class D uranium. The facility annual administrative limit for the Total Effective Dose Equivalent (TEDE) is 10 mSv (1000 mrem). Internal doses are evaluated at least annually.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to proposed intake limits on soluble uranium and the 10 mSv (1000 mrem) TEDE and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.

Continuous air monitoring in Airborne Radioactivity Areas may be performed to complement the bioassay program. Alarm setpoints on the continuous air monitors in the Airborne Radioactivity Areas may be used to provide an indication that internal exposures may be approaching the action limit.

If the facility annual administrative limit is exceeded as determined from bioassay results, then an investigation is performed and documented to determine what types of activities may have contributed to the worker's internal exposure. The action limit is based on ALARA principles. Other factors such as the biological elimination of uranium are considered. This investigation may include, but is not limited to procedural reviews, efficiency studies of the air handling system, and work practices.

4.7.7 Evaluation of Doses

Dose evaluations may be performed at more frequent intervals and should be performed when reasonable suspicion exists regarding an abnormal exposure. The internal and external exposure values are summed in accordance with 10 CFR 20.1202 (CFR, 2003r). Procedures for the evaluation and summation of doses are based on the guidance contained in Regulatory Guides 8.7 (NRC, 1992a) and 8.34 (NRC, 1992c).

4.7.8 Monitor Stations

Monitor stations are the entry and exit points for Restricted Areas. Monitors are provided to detect radioactive contamination on personnel and their personal items, including hard hats. All personnel are required to monitor themselves, any hand-carried personal items, and hard hats prior to exiting a Restricted Area. Radiation protection management is responsible for Monitor Station provision and maintenance. Figure 4.7-2, Projected Radiological Zones shows the anticipated Restricted Areas. Monitor Station locations are evaluated and moved as necessary in response to changes in the facility radiological conditions.

4.7.9 Locker Rooms

Locker rooms for men and women are provided for personnel to change into appropriate work clothing and store personal belongings. The following facilities are provided for in the locker room area:

- Shower Rooms - shower rooms for men and women are provided as a place for personnel to wash/clean up after work. These shower rooms are not intended for personnel decontamination.
- Restrooms - restrooms for men and women are provided. These rooms are not for personnel decontamination.
- First Aid Station - a first aid station is provided to treat injured personnel.
- Personnel Decontamination Area - a personnel decontamination area is provided to handle cases of accidental radioactive contamination. A handwashing sink and a shower are provided for contamination removal.
- Information Area - an information area is provided to notify personnel of information important to radiation protection.

4.7.10 Storage Areas

Storage areas are provided for the following items:

- Protective (i.e., anti-contamination) clothing
- Respiratory protection equipment
- Shower rooms supplies
- Radiation protection supplies.

4.8 CONTAMINATION AND RADIATION CONTROL

The goal of maintaining occupational internal and external radiation exposures ALARA encompasses the individual's dose as well as the collective dose of the entire working population. Since the total effective dose equivalent (TEDE) is the sum of the internal and external exposures, the Radiation Protection Program addresses both contamination control and external radiation protection.

Listed below are examples of design and operating considerations that are implemented at the facility to reduce personnel radiation exposures:

- The enrichment process, with the exception of the Liquid Sampling part, is maintained under sub atmospheric pressure. The constant containment of UF_6 precludes direct contact with radioactive materials by personnel.

- Self-monitoring is required upon exit from Restricted Areas. Personnel are required to notify a member of the radiation protection staff if contamination is detected.

- All personnel are trained in emergency evacuation procedures in accordance with the facility Emergency Plan.

- Air flow rates at exhausted enclosures and close-capture points, when in use, are adequate to preclude escape of airborne uranium and minimize the potential for intake by workers. Air flow rates are checked monthly when in use and after modification of any hood, exhausted enclosure, close-capture point equipment or ventilation system serving these barriers.

4.8.1 Internal Exposures

Because the radionuclides present in this facility under routine operations are primarily alpha and beta emitters (with some low-energy gamma rays), the potential for significant internal exposure is greater than that for external exposure. Parameters important to determining internal doses are:

- The quantity of radioactive material taken into the body

- The chemical form of the radioactive material

- The type and half-life of radionuclide involved

- The time interval over which the material remains in the body.

The principal modes by which radioactive material can be taken into the body are:

- Inhalation

- Ingestion

- Absorption through the skin
- Injection through wounds.

4.8.1.1 Bioassay

Internal radiological exposures are evaluated annually as noted in Section 4.7.7, Evaluation of Doses. Based on the results of air sample monitoring data, bioassays are performed for all personnel who are likely to have had an intake of one milligram of uranium during a week. This is 10% of the 10 mg (3.5 E-4 oz) in a week regulatory limit (10 CFR 20.1201(e) (CFR, 2003f)) for intake of Class D uranium. The bioassay program has a sensitivity of 5 μg/L (7 E-7 oz/gal) of uranium concentration, assuming that the sample is taken within ten days of the postulated intake and that at least 1.4 L (0.37 gal) of sample is available from a 24-hour sampling period. Until urinalysis results indicate less than 15 μg/L (2.0 E-6 oz/gal) of uranium concentration, workers are restricted from activities that could routinely or accidentally result in internal exposures to soluble uranium.

It might not be possible to achieve a sensitivity of 5 μg/L (7 E-7 oz/gal); if for example, all reasonable attempts to obtain a 1.4 L (0.37 gal) 24-hour sample within 10 days fail. In such a case, the sample is analyzed for uranium concentration (if measurable) and the worker's intake is estimated using other available data.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the internal bioassay program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.2.

4.8.1.2 Air Monitoring and Sampling

Airborne activity in work areas is regularly determined in accordance with written procedures. Continuous air sampling in airborne radioactivity areas may be performed to complement the bioassay program. Using the values specified in 10 CFR 20 Appendix B (CFR, 2003m), if a worker could have inhaled radionuclide concentrations that are likely to exceed 12 DAC-hours in one week (seven days), then bioassay is conducted within 72 hours after the suspected or known exposure. Follow-up bioassay measurements are conducted to determine the committed effective dose equivalent. Until urinalysis results indicate less than 15 micrograms per liter uranium concentration, workers are restricted from activities that could routinely or accidentally result in internal exposures to soluble uranium.

Active on-line monitors for airborne alpha emitters are used to measure representative airborne concentrations of radionuclides that may be due to facility operation. On-line monitoring for gross alpha activity is performed assuming all the alpha activity is due to uranium. When airborne activity data is used for dose calculations, the assumption is that all the activity is due

to ^{234}U, class D material. The lower limit of detection is either 0.02 mg (7.16 E-7 oz) of uranium in the total sample or 3.7 nBq/mL (1 E-13 μCi/mL) gross alpha concentration. An action level is established at 1 mg (3.53 E-5 oz) of total uranium likely to be inhaled by a worker in seven days.

Monitors are permanently located in Restricted Areas. These permanent monitors are operated to collect continuous samples. When air sampling is conducted using continuous air sampling devices, the filters are changed and analyzed at the following frequencies:

- Weekly and following any indication of release that might lead to airborne concentrations of uranium that are likely to exceed (1) 10% of the values listed in 10 CFR 20.1003 (CFR, 2003n), or (2) the total uranium action level of one milligram of total uranium inhaled in one week.

- Each Shift, following changes in process equipment or process control, and following detection of any event (e.g., leakage, spillage or blockage of process equipment) that are likely to exceed (1) 10% of the values listed in 10 CFR 20.1003 (CFR, 2003n), Airborne Radioactivity Area, or (2) the total uranium action level of one milligram inhaled by a worker in one week.

The representativeness of the workstation air samplers shall be checked annually and when significant process or equipment changes have been made. Facility procedures specify how representativeness is determined.

Plant areas surveyed as described in this section include as a minimum UF_6 processing areas, decontamination areas, waste processing areas and laboratories. Continuous air monitors (e.g., stationary samplers or personnel lapel samplers) may be substituted when appropriate, as when continuous monitoring may not be reasonably achieved.

Action levels are based on trending of data collected during facility operation. Investigations are performed if airborne activity:

A. Exceeds 10% of the values listed in 10 CFR 20.1003 (CFR, 2003n) for Airborne Radioactivity Areas

B. Shows a short-term increase of a factor of 10 over historical data from the previous 12 months.

Corrective actions include investigation of the adverse trend and an evaluation of the need for changes, consistent with the principles of ALARA.

4.8.2 External Exposures

As noted previously, the potential for significant external exposure to personnel under routine operating conditions is less significant than that for internal exposures. This is primarily due to the nature of the radionuclides present in the facility.

Parameters important in determining dose from external exposures are:

- The length of time the worker remains in the radiation field
- The intensity of the radiation field
- The portion of the body receiving the dose.

Historical data from European facilities of similar construction show relatively low doses compared to nuclear power plant doses.

4.8.3 Procedures

Procedures are provided in the following areas to administratively control personnel radiation exposure:

- Operation
- Design
- Maintenance
- Modification
- Decontamination
- Surveillance
- Procurement.

4.8.4 Instrumentation

Two basic types of personnel monitoring equipment are used at the facility. These are count rate meters (as known as "friskers") and hand/foot monitors.

4.8.4.1 Friskers

These typically consist of a hand-held Eberline HP 210/260 (or equivalent) probe connected to a RM-14 (or equivalent) count rate meter. Instructions for the use of these instruments are posted in a prominent location near the instrument. Hand held friskers are typically placed in locations where conditions restrict the use of other monitors or for short-term use as necessary to ensure effective control of the spread of contamination.

4.8.4.2 Hand and Foot Monitors

These typically consist of multiple detectors arranged to monitor only hands and feet. Instructions for the use of these monitors are prominently posted on or near the instrument. Hand and foot monitors are used in applications where "pass-throughs" are frequent and where hand and foot monitoring is the major requirement. Portal monitors, that can quickly scan large surface areas of the body, may be used where the number of personnel exiting an area, available space, etc., makes their use advantageous.

4.8.5 Contamination Control

Small contamination areas (i.e., less than one-fourth of the room) may be roped off or otherwise segregated from the rest of a Restricted Area. Appropriate clothing and/or other equipment is used to minimize exposure to radioactive material and prevent the spread of contamination. Provisions for monitoring contamination and airborne activity levels are discussed below. A contamination monitor (frisker), a step-off pad and a container for any discarded protective clothing may be placed at the access/egress point to the work area. The entire Restricted Area is not posted as a Contaminated Area.

4.8.5.1 Surface Contamination

Contamination survey monitoring is performed for all UF_6 process areas. Surveys include routine checks of non-UF_6 process areas, including areas normally not contaminated. Monitoring includes direct radiation and removable contamination measurements. Survey procedures are based on the potential for contamination of an area and operational experience. The Restricted Areas are surveyed at least weekly. The lunch room and change rooms are surveyed at least daily.

Removable surface contamination is considered uranium contamination that is present on a surface and that can be transferred to a dry smear paper by rubbing with moderate pressure. The facility uses various instruments such as proportional counters, alpha scintillation counters and thin window Geiger-Mueller tubes, to evaluate contamination levels.

Laundered protective clothing is periodically surveyed for gross alpha and gross beta contamination. Levels of less than 2.5 Bq/100 cm^2 (150 dpm/100 cm^2), alpha or beta/gamma are acceptable. This action level should be readily achievable since most of the radioactive material that can contaminate protective clothing at the facility is in soluble form and is easily removed by laundering.

If surface contamination levels exceed the following levels, clean-up of the contamination is initiated within 24 hours of the completion of the analysis:

- Removable contamination: 83.3 Bq/100 cm^2 (5000 dpm/100 cm^2) alpha or beta/gamma
- Fixed contamination: 4.2 kBq/100 cm^2 (250,000 dpm/100 cm^2) alpha or beta/gamma

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the surface and personnel contamination control program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.4.

4.9 MAINTENANCE AREAS-METHODS AND PROCEDURES FOR CONTAMINATION CONTROL

Designing processes and equipment that contain radioactive material to require as little maintenance as possible ensures that personnel radiation exposures are ALARA. Additional exposure reductions are achieved by:

A. Removing as much radioactive material as possible from the equipment and the area prior to maintenance, thereby reducing the intensity of the radiation field

B. Providing adequate space for ease of maintenance reducing the length of time required to complete the task, thereby reducing the time of exposure

C. Preparing and using procedures that contain specifications for tools and equipment needed to complete the job

D. Proper job planning, including practice on mockups

E. Previews of previous similar jobs

F. Identification and communication of the highest contamination areas to the workers prior to the start of work.

4.9.1 Decontamination Workshop

The Contaminated Workshop and Decontamination System are located in the same room in the TSB. This room is called the Decontamination Workshop. The Decontamination Workshop in the TSB contains an area to break down and strip contaminated equipment and to decontaminate the equipment and its components. The decontamination systems in the workshop are designed to remove radioactive contamination from contaminated materials and equipment. The only significant forms of radioactive contamination found in the facility are uranium hexafluoride (UF_6), uranium tetrafluoride (UF_4) and uranyl fluoride (UO_2F_2).

One of the functions of the Decontamination Workshop is to provide a maintenance facility for both UF_6 pumps and for vacuum pumps. The workshop is used for the temporary storage and subsequent dismantling of failed pumps. The dismantling area is in physical proximity to the decontamination train, in which the dismantled pump components are processed.

The process carried out within the Decontamination Workshop begins with receipt and storage of contaminated pumps, out-gassing, Fomblin oil removal and storage, and pump stripping. Activities for the dismantling and maintenance of other plant components are also carried out. Other components commonly decontaminated besides pumps include valves, piping, instruments, sample bottles, tools, and scrap metal. Personnel entry into the facility is via a sub-change facility. This area has the required contamination area access controls, washing and monitoring facilities.

The decontamination part of the process consists of a series of steps following equipment disassembly including degreasing, decontamination, drying, and inspection. Items from uranium hexafluoride systems, waste handling systems, and miscellaneous other items are decontaminated in this system.

4.9.2 Laundry System

The Laundry System cleans contaminated and soiled clothing and other articles which have been used throughout the plant. It contains the resulting solid and liquid wastes for transfer to appropriate treatment and disposal facilities. The Laundry System receives the clothing and articles from the plant in plastic bin bags, taken from containers strategically positioned within the plant. Clean clothing and articles are delivered to storage areas located within the plant. The Laundry System components are located in the Laundry room of the TSB.

The Laundry System collects, sorts, cleans, dries, and inspects clothing and articles used in Restricted Areas of the plant. Laundry collection is divided into two main groups; articles with a low probability of contamination and articles with a high probability of contamination. Those articles unlikely to have been contaminated are further sorted into lightly soiled and heavily soiled groups. The sorting is done on a table underneath a vent hood that is connected to the TSB GEVS. All lightly soiled articles are cleaned in the laundry. Heavily soiled articles are inspected and any considered to be difficult to clean (i.e., those with significant amounts of grease or oil on them) are transferred to the Solid Waste Collection System without cleaning. Articles from one plant department are not cleaned with articles from another plant department.

Special water-absorbent bags are used to collect the articles that are more likely to be contaminated. These articles may include pressure suits and items worn when, for example, it is required to disconnect or “open up” an existing plant system. These articles that are more likely to be contaminated are cleaned separately. Expected contaminants on the laundry include slight amounts of uranyl fluoride (UO_2F_2) and uranium tetrafluoride (UF_4).

When sorting is completed, the articles are placed in a washing machine in batches. No “dry cleaning” solvents are used. Wastewater from the washing machine is discharged to one of three Laundry Effluent Monitor Tanks in the Liquid Effluent Collection and Treatment System. The laundry effluent is then sampled, analyzed, and transferred to the Treated Effluent Evaporative Basin or to the Precipitation Treatment Tank for additional treatment as necessary.

When the washing cycle is complete, the wet laundry is placed in an electrically heated dryer. The dryer has variable temperature settings, and the hot wet air is exhausted to the atmosphere through a lint drawer that is built into the dryer. The lint from the drawer is then sent to the Solid Waste Collection System as combustible waste. Dry laundry is removed from the dryer and placed on the laundry inspection table for inspection and folding. Folded laundry is returned to storage areas in the plant.

4.10 DECONTAMINATION POLICY AND PROVISIONS

Removing radioactive material from equipment, to the extent reasonably possible prior to servicing, reduces exposures to personnel who work around and service contaminated equipment. Surface contamination is removed to minimize its spread to other areas of the facility. Surfaces such as floors and walls are designed to be smooth, nonporous and free of cracks so that they can be more easily decontaminated.

Decontamination facilities and procedures for the Technical Services Building and the Separations Building Module have been discussed above. For the remaining areas of the Separations Building Module, decontamination requirements involve only localized clean-up at areas where maintenance has been or is being performed that involves opening a uranium-containing system. All decontamination of components removed from their systems for maintenance is performed in Technical Services Building. No other areas of the facility normally require decontamination.

The facility follows NRC Branch Technical Position: Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material (NRC, 1993e). This guide applies to the abandonment or release for unrestricted use, of surfaces, premises and equipment.

4.11 ADDITIONAL PROGRAM COMMITMENTS

The following section describes additional program commitments related to the Radiation Protection Program.

4.11.1 Leak-Testing Byproduct Material Sources

In addition to the uranium processed at the facility, other sources of radioactivity are used. These sources are small calibration sources used for instrument calibration and response checking. These byproduct material sources may be in solid, liquid, or gaseous form; the sources may be sealed or unsealed. Both types of sources present a small radiation exposure risk to facility workers. Typical byproduct material quantities and uses for a Urenco uranium enrichment centrifuge plant are summarized in Table 4.11-1, Typical Quantities of Byproduct Material for a Urenco Uranium Enrichment Centrifuge Plant. The byproduct materials for the NEF will be identified during the design phase and the Safety Analysis Report will be revised accordingly. Leak-testing of sources is performed in accordance with the following NRC Branch Technical Positions (BTPs):

A. License Condition for Leak-Testing Sealed Byproduct Material Sources (NRC,1993b)

B. License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters (NRC, 1993c)

C. License Condition for Leak-Testing Sealed Uranium Sources (NRC, 1993d)

The following BTPs were not included in this section since the facility has not requested sources containing plutonium (refer to Table 4.11-1):

- *License Condition for Leak-Testing Sealed Plutonium Sources*, April 1993
- *License Condition for Plutonium Alpha Sources,* April 1993.

4.11.2 Records and Reports

The facility meets the following regulations for the additional program commitments applicable to records and reports:

- 10 CFR 20 (CFR, 2003b), Subpart L (Records), Subpart M (Reports)
- Section 70.61 (Performance requirements) (CFR, 2003e)
- Section 70.74 (Additional reporting requirements) (CFR, 2003s).

The facility Records Management program is described in Section 11.7, Records Management. The facility maintains complete records of the Radiation Protection Program for at least the life of the facility.

The facility maintains records of the radiation protection program (including program provisions, audits, and reviews of the program content and implementation), radiation survey results (air sampling, bioassays, external-exposure data from monitoring of individuals, internal intakes of radioactive material), and results of corrective action program referrals, RWPs and planned special exposures.

By procedure, the facility will report to the NRC, within the time specified in 10 CFR 20.2202 (CFR, 2003t) and 10 CFR 70.74 (CFR, 2003s), any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR 20 (CFR, 2003b). The facility will prepare and submit to the NRC an annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b) (CFR, 2003u).

As previously noted in this chapter, LES will refer to the facility's corrective action program any radiation incident that results in an occupational exposure that exceeds the dose limits in 10 CFR 20, Appendix B (CFR, 2003m), or is required to be reported per 10 CFR 70.74 (CFR, 2003s). The facility reports to the NRC both the corrective action taken (or planned) to protect against a recurrence and the proposed schedule to achieve compliance with the applicable license condition or conditions.

4.12 REFERENCES

ANSI, 1978. Radiation Protection Instrumentation Test and Calibration, ANSI N323-1978, American National Standards Institute, 1978.

ANSI, 1980. Testing of Nuclear Air Cleaning Systems, ANSI N510-1980, American National Standards Institute, 1980.

ANSI, 1981. Performance Requirements for Pocket-Sized Alarm Dosimeters and Alarm Ratemeters, ANSI N13.27-1981, American National Standards Institute, 1981.

ANSI, 1983. Dosimetry - Personnel Dosimetry Performance-Criteria for Testing, ANSI N13.11-1983, American National Standards Institute, 1983.

ANSI, 1985. Radiation Detectors—Personnel Thermoluminescence Dosimetry Systems-Performance, ANSI N13.15-1985, American National Standards Institute, 1985.

ANSI, 1989. Practice for Occupational Radiation Exposure Records Systems, ANSI N13.6-1966 (R1989), American National Standards Institute, 1989.

ANSI, 1992. Practices for Respiratory Protection, ANSI Z88.2-1992, American National Standards Institute, 1992.

ANSI, 1993a. Selection, Qualification and Training of Personnel for Nuclear Power Plants, ANSI 3.1-1993, American National Standards Institute, 1993.

ANSI, 1993b. Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities, ANSI N13.1-1969 (R1993), American National Standards Institute, 1993.

ANSI, 1995. Bioassay Program for Uranium, ANSI/HPS N13.22-1995, American National Standards Institute, 1995.

ANSI, 1996. Performance Criteria for Radiobioassay, ANSI/HPS N13.30, 1996, American National Standards Institute, 1996.

ASTM, 1989. Developing Training Programs in the Nuclear Fuel Cycle, ASTM C986-89, American Society for Testing and Materials, 1989.

ASTM, 1995. Radiological Protection Training for Nuclear Facility Workers, ASTM E1168-95, American Society for Testing and Materials, 1995.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 19, Notices, Instructions, and Reports to Workers: Inspections and Investigations, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Part 20, Standards for Protection Against Radiation, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 20.1101, Radiation protection programs, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Section 20.1201, Occupational dose limits for adults, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Section 20.1208, Dose equivalent to an embryo/fetus, 2003.

CFR, 2003h. Title 10, Code of Federal Regulations, Section 70.22, Contents of applications, 2003.

CFR, 2003i. Title 10, Code of Federal Regulations, Section 19.12, Instructions to workers, 2003.

CFR, 2003j. Title 10, Code of Federal Regulations, Section 20.2110, Form of records, 2003.

CFR, 2003k. Title 10, Code of Federal Regulations, Section 19.13, Notifications and reports to individuals, 2003.

CFR, 2003l. Title 29, Code of Federal Regulations, Part 1910, Occupational Safety and Health Standards, 2003.

CFR, 2003m. Title 10, Code of Federal Regulations, Part 20, Appendix B, Annual Limits on Intakes (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage, 2003.

CFR, 2003n. Title 10, Code of Federal Regulations, Section 20.1003, Definitions, 2003.

CFR, 2003o. Title 10, Code of Federal Regulations, Section 20.1301, Dose limits for individual members of the public, 2003.

CFR, 2003p. Title 40, Code of Federal Regulations, Part 190, Environmental Radiation Protection Standard For Nuclear Power Operations, 2003.

CFR, 2003q. Title 10, Code of Federal Regulation, Section 20.1902, Posting requirements, 2003.

CFR, 2003r. Title 10, Code of Federal Regulations, Section 20.1202, Compliance with requirements for summation of external and internal does, 2003.

CFR, 2003s. Title 10, Code of Federal Regulations, Section 70.74, Additional reporting requirements, 2003.

CFR, 2003t. Title 10, Code of Federal Regulations, Section 20.2202, Method for obtaining approval of proposed disposal procedures, 2003.

CFR, 2003u. Title 10, Code of Federal Regulations, Section 20.2206, Transfer for disposal and manifests, 2003.

CGA, 1997. Commodity Specification for Air, Publication G-7.1, Compressed Gas Association, 1997.

ERDA, 1976. Nuclear Air Cleaning Handbook, ERDA 76-21, Energy Research and Development Administration, 1976.

LES, 1993. Claiborne Enrichment Center Safety Analysis Report, Louisiana Energy Services, December 1993.

NCRP, 1978. Operational Radiation Safety Program, NCRP Report No. 59, National Council on Radiation Protection and Measurements, 1978.

NRC, 1973a. Guide for Administrative Practice in Radiation Monitoring, Regulatory Guide 8.2, U.S. Nuclear Regulatory Commission, February 1973.

NRC, 1973b. Direct-Reading and Indirect-Reading Pocket Dosimeters, Regulatory Guide 8.4, U.S. Nuclear Regulatory Commission, February 1973.

NRC, 1977. Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable, Regulatory Guide 8.10 Rev. 1-R, U.S. Nuclear Regulatory Commission, May 1977.

NRC, 1979. Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication, Regulatory Guide 8.24, Rev. 1, U.S. Nuclear Regulatory Commission, October 1979,

NRC, 1992a. Instructions for Recording and Reporting Occupational Radiation Exposure Data, Regulatory Guide 8.7, Rev.1, U.S. Nuclear Regulatory Commission, June 1992.

NRC, 1992b. Air Sampling in the Workplace, Regulatory Guide 8.25, Rev. 1, U.S. Nuclear Regulatory Commission, June 1992.

NRC, 1992c. Monitoring Criteria and Methods To Calculate Occupational Radiation Doses, Regulatory Guide 8.34, U.S. Nuclear Regulatory Commission, July 1992.

NRC, 1993a. Air Sampling in the Workplace, NUREG-1400, U.S. Nuclear Regulatory Commission, September 1993.

NRC, 1993b. License Condition for Leak-Testing Sealed Byproduct Material Sources, Branch Technical Position, U.S. Nuclear Regulatory Commission, April 1993.

NRC, 1993c. License Condition for Leak-Testing Sealed Sources Which Contain Alpha and/or Beta-Gamma Emitters, Branch Technical Position, U.S. Nuclear Regulatory Commission, April 1993.

NRC, 1993d. License Condition for Leak-Testing Sealed Uranium Sources, Branch Technical Position, U.S. Nuclear Regulatory Commission, April 1993.

NRC, 1993e. Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct Source or Special Nuclear Material, Branch Technical Position, U.S. Nuclear Regulatory Commission, April 1993.

NRC, 1993f. Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program, Regulatory Guide 8.9, Rev. 1, U.S. Nuclear Regulatory Commission, July 1993.

NRC, 1993g. ALARA Levels for Effluents from Materials Facilities, Regulatory Guide 8.37, U.S. Nuclear Regulatory Commission, July 1993.

NRC, 1994. Safety Evaluation Report for the Claiborne Enrichment Center, Homer, Louisiana, NUREG-1491, U.S. Nuclear Regulatory Commission, January 1994.

NRC, 1996. Instructions Concerning Risks from Occupational Radiation Exposure, Regulatory Guide 8.29, U.S. Nuclear Regulatory Commission, February 1996.

NRC, 1999a. Instruction Concerning Prenatal Radiation Exposure, Regulatory Guide 8.13, Rev. 3; U.S. Nuclear Regulatory Commission, June 1999.

NRC, 1999b. Acceptable Programs for Respiratory Protection Regulatory Guide 8.15, U.S. Nuclear Regulatory Commission, October 1999.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

Urenco, 2000. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2000.

Urenco, 2001. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2001.

Urenco, 2002. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2002.

TABLES

Table 4.1-1 Administrative Radiation Exposure Limits

Page 1 of 1

	Administrative Limit
Total Effective Dose Equivalent (TEDE)	10 mSv/yr (1000 mrem/yr)

Notes:

a) Excludes accident situations

b) No routine extremity or skin monitoring is required

c) TEDE is the sum of internal dose and external dose received during routine operations

d) NRC limit is 50 mSv/yr (5000 mrem/yr)

Table 4.1-2 Estimated Dose Rates

Page 1 of 1

Area or Component	Dose Rate, mSv/hr (mrem/hr)
Plant general area (excluding Separations Building Module)	< 1 E-4 (< 0.01)
Separations Building Module – Cascade Halls	5 E-4 (0.05)
Separations Building Module –UF_6 Handling Area & Process Services Area	1 E-3 (0.1)
Empty used UF_6 shipping cylinder	0.1 on contact (10.0) 0.01 at 1 m (1.0)
Full UF_6 shipping cylinder	0.05 on contact (5.0) 2 E-3 at 1 m (0.2)

Table 4.1-3 Estimated Individual Exposures

Position	Annual Dose [(a)] mSv (mrem)
General Office Staff	< 0.05 (< 5.0)
Typical Operations & Maintenance Technician	1 (100)
Typical Cylinder Handler	3 (300)

(a) The average worker exposure at the Urenco Capenhurst facility during the years 1998 through 2002 was approximately 0.2 mSv (20 mrem) (Urenco, 2000; Urenco, 2001; Urenco, 2002)

Table 4.7-1 Radiation Emitted from Natural UF_6 Feed

Page 1 of 1

Element	Nuclide Symbol	Half-Life	Maximum Radiation Energies (Mev) and Intensities		
			alpha (α)	beta (β)	gamma (γ)
92 uranium	^{238}U	4.5E+9 yr	4.15 25% 4.20 75%	none	0.013 8.8%
90 thorium	^{231}Th	26 hr	none	0.39 ~100%	0.025 14.7%
90 thorium	^{234}Th	24 d	none	0.19 73% 0.10 27%	0.06 3.8% 0.09 5.4%
91 protactinium	^{234}Pa	1.2 min	none	2.28 99%	0.766 0.21% 1.001 0.60%
92 uranium	^{234}U	2.5E+5 yr	4.72 28% 4.78 72%	none	0.053 0.12%
92 uranium	^{235}U	7.04E+8 yr	4.37 17% 4.40 55% 4.60 14%	none	0.143 12% 0.185 54% 0.205 6%

Text removed under 10 CFR 2.390.

FIGURES

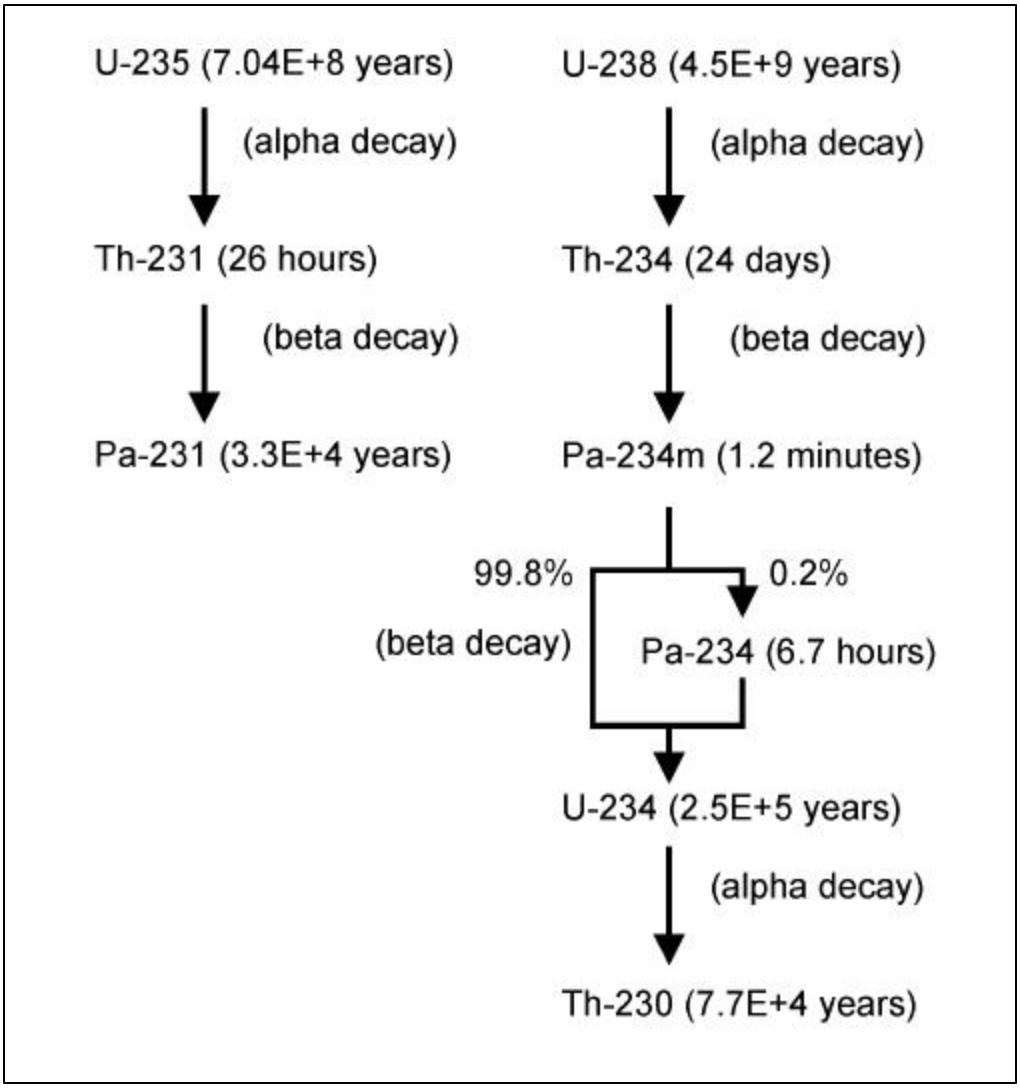


FIGURE 4.7-1

URANIUM AND DECAY PRODUCTS OF INTEREST

REFERENCE NUMBER
Figure 4.7—1.doc

REVISION DATE: DECEMBER 2003

Figure removed under 10 CFR 2.390.

TABLE OF CONTENTS

Page

5.0 NUCLEAR CRITICALITY SAFETY 5.0-1

5.1 THE NUCLEAR CRITICALITY SAFETY (NCS) PROGRAM 5.1-1

5.1.1 Management of the Nuclear Criticality Safety (NCS) Program 5.1-1
5.1.2 Control Methods for Prevention of Criticality 5.1-3
5.1.3 Safe Margins Against Criticality 5.1-5
5.1.4 Description of Safety Criteria 5.1-6
5.1.5 Organization and Administration 5.1-6

5.2 METHODOLOGIES AND TECHNICAL PRACTICES 5.2-1

5.2.1 Methodology 5.2-1
5.2.1.1 Methods Validation 5.2-1
5.2.1.2 Limits on Control and Controlled Parameters 5.2-2
5.2.1.3 General Nuclear Criticality Safety Methodology 5.2-3
5.2.1.4 Nuclear Criticality Safety Analyses 5.2-4
5.2.1.5 Additional Nuclear Criticality Safety Analyses 5.2-5
5.2.1.6 Nuclear Criticality Safety Evaluations (NCSE) 5.2-6

5.3 NUCLEAR CRITICALITY SAFETY ANALYSES RESULTS 5.3-1

5.3.1 Centrifuges and Cascades 5.3-1
5.3.2 Product Cylinders 5.3-1
5.3.3 Product Vent Subsystem UF_6 Cold Traps 5.3-2
5.3.4 Product Vent Subsystem Pumping System 5.3-3
5.3.5 Contingency Dump Trap 5.3-4
5.3.6 Product Pumping Train UF_6 Pumps 5.3-4
5.3.7 Vacuum Cleaners 5.3-5
5.3.8 Technical Services Building Solid Waste Collection Room 5.3-6
5.3.9 Technical Services Building Ventilated Room 5.3-6
5.3.10 Technical Services Building Chemical Laboratory 5.3-7
5.3.11 Technical Services Building Decontamination Workshop 5.3-8
5.3.12 Technical Services Building Fomblin Oil Recovery System 5.3-9
5.3.13 Separations Building Gaseous Effluent Vent System 5.3-9
5.3.14 Technical Services Building Gaseous Effluent Vent System 5.3-10
5.3.15 UF_6 Product Pipework 5.3-11

5.4 CRITICALITY ACCIDENT ALARM SYSTEM (CAAS) 5.4-1

5.5 REPORTING 5.5-1

5.6 REFERENCES 5.6-1

LIST OF TABLES

Table 5.1-1 Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2
Table 5.1-2 Safety Criteria for Buildings/Systems/Components
Table 5.2-1 Uranium Solution Experiments Used for Validation

LIST OF FIGURES

Figure 5.2-1 Validation Results for Uranium Solutions

5.0 NUCLEAR CRITICALITY SAFETY

The Nuclear Criticality Safety Program for the National Enrichment Facility (NEF) is in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities (NRC, 1998). Regulatory Guide 3.71 (NRC, 1998) provides guidance on complying with the applicable portions of NRC regulations, including 10 CFR 70 (CFR, 2003a), by describing procedures for preventing nuclear criticality accidents in operations involving handling, processing, storing, and transporting special nuclear material (SNM) at fuel and material facilities. The facility is committed to following the guidelines in this regulatory guide for specific ANSI/ANS criticality safety standards with the exception of ANSI/ANS-8.9-1987, "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Material." Piping configurations containing aqueous solutions of fissile material will be evaluated in accordance with ANSI/ANS-8.1-1998, using validated methods to determine subcritical limits.

The information provided in this chapter, the corresponding regulatory requirements, and the section of NUREG-1520 (NRC, 2002), Chapter 5 in which the NRC acceptance criteria are presented is summarized below.

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 5 Reference
Section 5.1 Nuclear Criticality Safety (NCS) Program		
Management of the NCS Program	70.61(d) 70.64(a)	5.4.3.1
Control Methods for Prevention of Criticality	70.61	5.4.3.4.2
Safe Margins Against Criticality	70.61	5.4.3.4.2
Description of Safety Criteria	70.61	5.4.3.4.2
Organization and Administration	70.61	5.4.3.2
Section 5.2 Methodologies and Technical Practices		
Methodology	70.61	5.4.3.4.1 5.4.3.4.4 5.4.3.4.6
Section 5.3 Nuclear Criticality Safety Analyses Results		
Criticality analyses results for various systems and configurations	70.61(d)	5.4.3.4.4
Section 5.4 Criticality Accident Alarm System (CAAS)		
Criticality Accident Alarm System	70.24	5.4.3.4.3
Section 5.5 Reporting		
Reporting Requirements	Appendix A	5.4.3.4.7 (7)

5.1 THE NUCLEAR CRITICALITY SAFETY (NCS) PROGRAM

The facility has been designed and will be constructed and operated such that a nuclear criticality event is prevented, and to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a). Nuclear criticality safety at the facility is assured by designing the facility, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and any credible accident. Items Relied On For Safety (IROFS) identified to ensure subcriticality are discussed in Section 3.8, Items Relied on for Safety (IROFS).

5.1.1 Management of the Nuclear Criticality Safety (NCS) Program

The NCS criteria in Section 5.2, Methodologies and Technical Practices, are used for managing criticality safety and include adherence to the double contingency principle as stated in the ANSI/ANS-8.1-1998, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). The adopted double contingency principle states "process design should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." Each process that has accident sequences that could result in an inadvertent nuclear criticality at the NEF has double contingency protection.

In most cases, double contingency protection is provided by at least two-parameter control. Using these criteria, including the double contingency principle, low enriched uranium enrichment facilities have never had an accidental criticality. The plant will produce no greater than 5.0 $^{w}/_{o}$ enrichment. However, as additional conservatism, the nuclear criticality safety analyses are performed assuming a ^{235}U enrichment of 6.0 $^{w}/_{o}$, except for Contingency Dump System traps which are analyzed assuming a ^{235}U enrichment of 1.5 $^{w}/_{o}$, and include appropriate margins to safety. In accordance with 10 CFR 70.61(d) (CFR, 2003b), the general criticality safety philosophy is to prevent accidental uranium enrichment excesses, provide geometrical safety when practical, provide for moderation controls within the UF_6 processes and impose strict mass limits on containers of aqueous, solvent based, or acid solutions containing uranium. Interaction controls provide for safe movement and storage of components. Plant and equipment features assure prevention of excessive enrichment. The plant is divided into six distinctly separate Assay Units (called Cascade Halls) with no common UF_6 piping. UF_6 blending is done in a physically separate portion of the plant. Process piping, individual centrifuges and chemical traps other than the contingency dump chemical traps, are safe by limits placed on their diameters. Product cylinders rely upon uranium enrichment, moderation control and mass limits to protect against the possibility of a criticality event. Each of the liquid effluent collection tanks that hold uranium in solution is mass controlled, as none are geometrically safe. As required by 10 CFR 70.64(a) (CFR, 2003c), by observing the double contingency principle throughout the plant, a criticality accident is prevented. In addition to the double contingency principle, effective management of the NCS Program includes:

- An NCS program to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a) will be developed, implemented, and maintained.

- Safety parameters and procedures will be established.

- The NCS program structure, including definition of the responsibilities and authorities of key program personnel will be provided.

- The NCS methodologies and technical practices will be kept applicable to current configuration by means of the configuration management function. The NCS program will be upgraded, as necessary, to reflect changes in the ISA or NCS methodologies and to modify operating and maintenance procedures in ways that could reduce the likelihood of occurrence of an inadvertent nuclear criticality.

- The NCS program will be used to establish and maintain NCS safety limits and NCS operating limits for IROFS in nuclear processes and a commitment to maintain adequate management measures to ensure the availability and reliability of the IROFS.

- NCS postings will be provided and maintained current.

- NCS emergency procedure training will be provided.

- The NCS baseline design criteria requirements in 10 CFR 70.64(a) (CFR, 2003c) will be adhered to. The NCS baseline design criteria are described in Table 5.1-2, Safety Criteria for Buildings/Systems/Components.

- The NCS program will be used to evaluate modifications to operations, to recommend process parameter changes necessary to maintain the safe operation of the facility, and to select appropriate IROFS and management measures.

- The NCS program will be used to promptly detect NCS deficiencies by means of operational inspections, audits, and investigations. Deficiencies will be entered into the corrective action program so as to prevent recurrence of unacceptable performance deficiencies in IROFS, NCS function or management measures.

- NCS program records will be retained as described in Section 11.7, Records Management.

Training will be provided to individuals who handle nuclear material at the facility in criticality safety. The training is based upon the training program described in ANSI/ANS-8.20-1991, Nuclear Criticality Safety Training (ANSI, 1991). The training program is developed and implemented with input from the criticality safety staff, training staff, and management. The training focuses on the following:

- Appreciation of the physics of nuclear criticality safety.
- Analysis of jobs and tasks to determine what a worker must know to perform tasks efficiently.
- Design and development of learning objectives based upon the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker.
- Implementation of revised or temporary operating procedures.

Additional discussion of management measures is provided in Chapter 11, Management Measures.

5.1.2 Control Methods for Prevention of Criticality

The major controlling parameters used in the facility are enrichment control, geometry control, moderation control, and/or limitations on the mass as a function of enrichment. In addition, reflection, interaction, and heterogeneous effects are important parameters considered and applied where appropriate in nuclear criticality safety analyses. Nuclear Criticality Safety Evaluations and Analyses are used to identify the significant parameters affected within a particular system. All assumptions relating to process, equipment, material function, and operation, including credible abnormal conditions, are justified, documented, and independently reviewed. Where possible, passive engineered controls are used to ensure NCS. The determination of the safe values of the major controlling parameters used to control criticality in the facility is described below.

Moderation control is in accordance with ANSI/ANS-8.22-1997, Nuclear Criticality Safety Based on Limiting and Controlling Moderators (ANSI, 1997). However, for the purposes of the criticality analyses, it is assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2 as described in Section 5.2.1.3.3, Uranium Accumulation and Moderation Assumption. A uniform aqueous solution of UO_2F_2, and a fixed enrichment are conservatively modeled using MONK8A (SA, 2001) and the JEF2.2 library. Criticality analyses were performed to determine the maximum value of a parameter to yield $k_{eff} = 1$. The criticality analyses were then repeated to determine the maximum value of the parameter to yield a $k_{eff} = 0.95$. Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, shows both the critical and safe limits for 5.0 $^{w}/_{o}$ and 6.0 $^{w}/_{o}$.

Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, lists the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, which are used as control parameters to prevent a nuclear criticality event. Although the NEF will be limited to 5.0 $^{w}/_{o}$ enrichment, as additional conservatism, the values in Table 5.1-2, Safety Criteria for Buildings/Systems/ Components, represent the limits based on 6.0 $^{w}/_{o}$ enrichment except for the Contingency Dump System traps which are limited to 1.5 $^{w}/_{o}$ ^{235}U.

The values on Table 5.1-1 are chosen to be critically safe when optimum light water moderation exists and reflection is considered within isolated systems. The conservative modeling techniques provide for more conservative values than provided in ANSI/ANS-8.1 (ANSI, 1998a). The product cylinders are only safe under conditions of limited moderation and enrichment. In such cases, both design and operating procedures are used to assure that these limits are not exceeded.

All Separation Plant components, which handle enriched UF_6, other than the Type 30B and 48Y cylinders and the first stage UF_6 pumps and contingency dump chemical traps, are safe by geometry. Centrifuge array criticality is precluded by a probability argument with multiple operational procedure barriers. Total moderator or H/U ratio control as appropriate precludes product cylinder criticality.

In the Technical Services Building (TSB) criticality safety for uranium loaded liquids is ensured by limiting the mass of uranium in any single tank to less than or equal to 12.2 kg U (26.9 lb U). Individual liquid storage bottles are safe by volume. Interaction in storage arrays is accounted for.

Based on the criticality analyses, the control parameters applied to NEF are as follows:

Enrichment

Enrichment is controlled to limit the percent ^{235}U within any process, vessel, or container, except the contingency dump system, to a maximum enrichment of 5 $^{w}/_{o}$. The design of the contingency dump system controls enrichment to a limit of 1.5 $^{w}/_{o}$ ^{235}U. Although NEF is limited to a maximum enrichment of 5 $^{w}/_{o}$, as added conservatism nuclear criticality safety is analyzed using an enrichment of 6 $^{w}/_{o}$ ^{235}U.

Geometry/Volume

Geometry/volume control may be used to ensure criticality safety within specific process operations or vessels, and within storage containers.

The geometry/volume limits are chosen to ensure k_{eff} (k_{calc} + 3 σ_{calc}) $\leq$ 0.95.

The safe values of geometry/volume define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming 6 $^{w}/_{o}$ ^{235}U for safety margin.

Moderation

Water and oil are the moderators considered in NEF. At NEF the only system where moderation is used as a control parameter is in the product cylinders. Moderation control is established consistent with the guidelines of ANSI/ANS-8.22-1997 (ANSI, 1997) and incorporates the criteria below:

- Controls are established to limit the amount of moderation entering the cylinders.
- When moderation is the only parameter used for criticality control, the following additional criteria are applied. These controls assure that at least two independent controls would have to fail before a criticality accident is possible.
    - Two independent controls are utilized to verify cylinder moderator content.
    - These controls are established to monitor and limit uncontrolled moderator prior to returning a cylinder to production thereby limiting the amount of uncontrolled moderator from entering a system to an acceptable limit.
    - The evaluation of the cylinders under moderation control includes the establishment of limits for the ratio of maximum moderator-to-fissile material for both normal operating and credible abnormal conditions. This analysis has been supported by parametric studies.
- When moderation is not considered a control parameter, either optimum moderation or worst case H/U ratio is assumed when performing criticality safety analysis.

Mass

Mass control may be utilized to limit the quantity of uranium within specific process operations, vessels, or storage containers. Mass control may be used on its own or in combination with

other control methods. Analysis or sampling is employed to verify the mass of the material. Conservative administrative limits for each operation are specified in the operating procedures.

Whenever mass control is established for a container, records are maintained for mass transfers into and out of the container. Establishment of mass limits for a container involves consideration of potential moderation, reflection, geometry, spacing, and enrichment. The evaluation considers normal operations and credible abnormal conditions for determination of the operating mass limit for the container and for the definition of subsequent controls necessary to prevent reaching the safety limits. When only administrative controls are used for mass controlled systems, double batching is conservatively assumed in the analysis.

Reflection

Reflection is considered when performing Nuclear Criticality Safety Evaluations and Analyses. The possibility of full water reflection is considered but the layout of the NEF is a very open design and it is highly unlikely that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. In addition, neither automatic sprinkler nor standpipe and hose systems are provided in the TSB, Separation Buildings, Blending and Liquid Sampling, CRDB, CAB, and Centrifuge Post Mortem areas. Therefore, full water reflection of vessels has therefore been discounted. However, some select analyses have been performed using full reflection for conservatism. Partial reflection of 2.5 cm of water is assumed where limited moderating materials (including humans) may be present It is recognized that concrete can be a more efficient reflector than water; therefore, it is modeled in analyses where it is present. When moderation control is identified in the ISA Summary, it is established consistent with the guidelines of ANSI/ANS-8.22-1997 (ANSI, 1997).

Interaction

Nuclear criticality safety evaluations and analyses consider the potential effects of interaction. A non-interacting unit is defined as a unit that is spaced an approved distance from other units such that the multiplication of the subject unit is not increased. Units may be considered non-interacting when they are separated by more than 60 cm (23.6 inches).

If a unit is considered interacting, nuclear criticality safety analyses are performed. Individual unit multiplication and array interaction are evaluated using the Monte Carlo computer code MONK8A to ensure k_{eff} (k_{calc} + 3 σ_{calc}) < 0.95.

Concentration, Density and Neutron Absorbers

NEF does not use mass concentration, density, or neutron absorbers as a criticality control parameter.

5.1.3 Safe Margins Against Criticality

Process operations require establishment of criticality safety limits. The facility UF_6 systems involve mostly gaseous operations. These operations are carried out under reduced atmospheric conditions (vacuum) or at slightly elevated pressures not exceeding three atmospheres. It is highly unlikely that any size changes of process piping, cylinders, cold traps, or chemical traps under these conditions, would lead to a criticality situation because a volume or mass limit may be exceeded.

Within the Separations Building, significant accumulations of enriched UF_6 reside only in the Product Low Temperature Take-off Stations, Product Liquid Sampling Autoclaves, Product Blending System or the UF_6 cold traps. All these, except the UF_6 cold traps, contain the UF_6 in 30B and 48Y cylinders. All these significant accumulations are within enclosures protecting them from water ingress. The facility design has minimized the possibility of accidental moderation by eliminating direct water contact with these cylinders of accumulated UF_6. In addition, the facility's stringent procedural controls for enriching the UF_6 assure that it does not become unacceptably hydrogen moderated while in process. The plant's UF_6 systems operating procedures contain safeguards against loss of moderation control (ANSI, 1997). No neutron poisons are relied upon to assure criticality safety.

5.1.4 Description of Safety Criteria

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, are applied to the facility to prevent a nuclear criticality event. Although the NEF will be limited to 5.0 $^{w}/_{o}$ enrichment, as additional conservatism, the values in Table 5.1-2, represent the limits based on 6.0 $^{w}/_{o}$ enrichment.

Where there are significant in-process accumulations of enriched uranium as UF_6, the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

5.1.5 Organization and Administration

The criticality safety organization is responsible for implementing the Nuclear Criticality Safety Program. During the design phase, the criticality safety function is performed within the design engineering organization. The criticality safety function for operations is described in the following section.

The criticality safety organization reports to the Health, Safety, and Environment (HS&E) Manager as described in Chapter 2, Organization and Administration. The HS&E Manager is accountable for overall criticality safety of the facility, is administratively independent of production responsibilities, and has the authority to shut down potentially unsafe operations.

Designated responsibilities of the criticality safety staff include the following:

- Establish the Nuclear Criticality Safety Program, including design criteria, procedures, and training
- Provide criticality safety support for integrated safety analyses and configuration control
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters
- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs) (i.e., non-calculation engineering judgments regarding whether existing criticality safety analyses bound the issue being evaluated or whether new or revised safety analyses are required)

- Perform criticality safety analyses (i.e., calculations, write NCSEs, and approve proposed changes in process conditions on equipment involving fissionable material)
- Specify criticality safety control requirements and functionality
- Provide advice and counsel on criticality safety control measures, including review and approval of operating procedures
- Support emergency response planning and events
- Evaluate the effectiveness of the Nuclear Criticality Safety Program using audits and assessments
- Provide criticality safety postings that identify administrative controls for operators in applicable work areas.

The minimum qualifications for a criticality safety engineer are a Bachelor of Science (BS) or Bachelor of Arts (BA) degree in science or engineering with at least two years of nuclear industry experience in criticality safety. A criticality safety engineer must understand and have experience in the application and direction of criticality safety programs. The HS&E Manager has the authority and responsibility to assign and direct activities for the criticality safety staff. The criticality safety engineer is responsible for implementation of the NCS program. Criticality safety engineers will be provided in sufficient numbers to implement and support the operation of the NCS program.

The NEF implements the intent of the administrative practices for criticality safety, as contained in Section 4.1.1 of American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1-1998, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). A policy will be established whereby personnel shall report defective NCS conditions and perform actions only in accordance with written, approved procedures. Unless a specific procedure deals with the situation, personnel shall report defective NCS conditions and take no action until the situation has been evaluated and recovery procedures provided.

5.2 METHODOLOGIES AND TECHNICAL PRACTICES

This section describes the methodologies and technical practices used to perform the Nuclear Criticality Safety (NCS) analyses. The determination of the NCS controlled parameters and their application and the determination of the NCS limits on IROFS are also presented.

5.2.1 Methodology

MONK8A (SA, 2001) is a powerful Monte Carlo tool for nuclear criticality safety analysis. The advanced geometry modeling capability and detailed continuous energy collision modeling treatments provide realistic 3-dimensional models for an accurate simulation of neutronic behavior to provide the best estimate neutron multiplication factor, k-effective. Complex models can be simply set up and verified. Additionally, MONK8A (SA, 2001) has demonstrable accuracy over a wide range of applications and is distributed with a validation database comprising critical experiments covering uranium, plutonium and mixed systems over a wide range of moderation and reflection. The experiments selected are regarded as being representative of systems that are widely encountered in the nuclear industry, particularly with respect to chemical plant operations, transportation and storage. The validation database is subject to on-going review and enhancement. A categorization option is available in MONK8A (SA, 2001) to assist the criticality analyst in determining the type of system being assessed and provides a quick check that a calculation is adequately covered by validation cases.

5.2.1.1 Methods Validation

The validation process establishes method bias by comparing measured results from laboratory critical experiments to method-calculated results for the same systems. The verification and validation processes are controlled and documented. The validation establishes a method bias by correlating the results of critical experiments with results calculated for the same systems by the method being validated. Critical experiments are selected to be representative of the systems to be evaluated in specific design applications. The range of experimental conditions encompassed by a selected set of benchmark experiments establishes the area of applicability over which the calculated method bias is applicable. Benchmark experiments are selected that resemble as closely as practical the systems being evaluated in the design application.

The extensive validation database contains a number of solution experiments applicable to this application involving both low and high-enriched uranium. The MONK8A (SA, 2001) code with the JEF2.2 library was validated against these experiments which are provided in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (NEA, 2002) and Nuclear Science and Engineering (NSE, 1962). The experiments chosen are provided in Table 5.2-1, Uranium Solution Experiments Used for Validation, along with a brief description. The overall mean calculated value from the 80 configurations is 1.0017 ± 0.0005 (AREVA, 2004) and the results are shown in Figure 5.2-1,Validation Results for Uranium Solutions, plotted against H/U-fissile ratio. If only the 36 low-enriched solutions are considered, the mean calculated value is 1.0007 ± 0.0005.

MONK8A is distributed in ready-to-run executable form. This approach provides the user with a level of quality assurance consistent with the needs of safety analysis. The traceability from

source code to executable code is maintained by the code vendor. The MONK8A software package contains a set of validation analyses which can be used to support the specific applications. Since the source code is not available to the user, the executable code is identical to that used for the validation analyses. The criticality analyses presented in Section 5.3 were performed with MONK8A utilizing the validation provided by the code vendor.

In accordance with the guidance in NUREG-1520 (NRC, 2002), code validation for the specific application has been performed (AREVA, 2004). Specifically, the experiments provided in Table 5.2-1, Uranium Solution Experiments Used for Validation, were calculated and documented as part of the integrated safety analysis for the National Enrichment Facility. The MONK8A computer code and JEF2.2 library are within the scope of the Quality Assurance Program.

5.2.1.2 Limits on Control and Controlled Parameters

The validation process established a bias by comparing calculations to measured critical experiments. With the bias determined, an upper safety limit (USL) can be determined using the following equation from NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology (NRC, 2001):

$$USL = 1.0 + Bias - \sigma_{Bias} - \Delta_{SM} - \Delta_{AOA}$$

Where the critical experiments are assumed to have a k_{eff} of unity, and the bias was determined by comparison of calculation to experiment. From Section 5.2.1.1, Methods Validation, the bias is positive and since a positive bias may be non-conservative, the bias is set to zero. The σ_{Bias} from Section 5.2.1.1, Methods Validation is 0.0005 and a value of 0.05 is assigned to the subcritical margin, Δ_{SM}. The term Δ_{AOA} is an additional subcritical margin to account for extensions in the area of applicability. Since the experiments in the benchmark are representative of the application, the term Δ_{AOA} is set to zero. Thus, the USL becomes:

$$USL = 1 - 0.0005 - 0.05 = 0.9495$$

NUREG/CR-6698 (NRC, 2001) requires that the following condition be demonstrated for all normal and credible abnormal operating conditions:

$$k_{calc} + 2\,\sigma_{calc} < USL$$

In the NCS analysis, σ_{calc} is shown to be greater than σ_{Bias}; therefore, the NEF will be designed using the more conservative equation:

$$k_{eff} = k_{calc} + 3\,\sigma_{calc} < 0.95$$

Additionally, criticality safety in the NEF is ensured by use of geometry, volume, mass and moderation control. Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2 provides the safe values of geometry, volume and mass at 5.0 $^{w}/_{o}$ enrichment UO_2F_2 to ensure the USL is met. Moreover, Table 5.1-2, Safety Criteria for Buildings/Systems/Components, provides the additional conservatism used in the design of the NEF. All criticality safety analyses use an enrichment of 6.0 $^{w}/_{o}$ ^{235}U, except for Contingency Dump System traps which are analyzed using an enrichment of 1.5 $^{w}/_{o}$ ^{235}U, while the facility is limited to an enrichment of 5.0 $^{w}/_{o}$ ^{235}U. Details of the NEF criticality safety analyses are provided in Section 5.3, NCS Analyses Results.

5.2.1.3 General Nuclear Criticality Safety Methodology

The nuclear criticality safety analyses results presented in Section 5.3 provide values of k-effective (k_{eff}) to conservatively meet the upper safety limit. The following sections provide a description of the major assumptions used in the criticality analysis.

5.2.1.3.1 Reflection Assumption

The layout of the NEF is a very open design and it is not considered credible that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. Full water reflection of vessels has therefore been discounted. However, where appropriate, spurious reflection due to walls, fixtures, personnel, etc. has been accounted for by assuming 2.5 cm (1.0 in) of water reflection around vessels.

5.2.1.3.2 Enrichment Assumption

The NEF will operate with a 5.0 $^{w}/_{o}$ ^{235}U enrichment limit. However, the nuclear criticality safety calculations used an enrichment of 6.0 $^{w}/_{o}$ ^{235}U. This assumption provides additional conservatism for plant design.

5.2.1.3.3 Uranium Accumulation and Moderation Assumption

Most components that form part of the centrifuge plant or are connected to it assume that any accumulation of uranium is taken to be in the form of a uranyl fluoride/water mixture at a maximum H/U atomic ratio of 7 (exceptions are discussed in the appropriate portions of Section 5.3). The ratio is based on the assumption that significant quantities of moderated uranium could only accumulate by reaction between UF_6 and moisture in air leaking into the plant. Due to the high vacuum requirements of a centrifuge plant, in-leakage is controlled at very low levels and thus the H/U ratio of 7 represents an abnormal condition. The maximum H/U ratio of 7 for the uranyl fluoride-water mixture is derived as follows:

The stoichiometric reaction between UF_6 and water vapor in the presence of excess UF_6 can be represented by the equation:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound. Experimental studies (Lychev, 1990) suggest that solid hydrates of compositions $UO_2F_2 \cdot 1.5H_2O$ and $UO_2F_2 \cdot 2H_2O$ can form in the presence of water vapor, the former composition being the stable form on exposure to atmosphere.

It is assumed that the hydrate $UO_2F_2 \cdot 1.5H_2O$ is formed and, additionally, that the hydrogen fluoride (HF) produced by the UF_6/water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:

$$UF_6 + 3.5H_2O \rightarrow UO_2F_2 \cdot 4HF \cdot 1.5H_2O$$

For the MONK8A (SA, 2001) calculations, the composition of the breakdown product was simplified to $UO_2F_2 \cdot 3.5H_2O$ that gives the same H/U ratio of 7 as above.

In the case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated perfluorinated polyether (PFPE) type lubricant, often referred to by the trade name “Fomblin.” Mixtures of UF_6

and PFPE oil would be a less conservative case than a uranyl fluoride/water mixture, since the maximum HF solubility in PFPE is only about 0.1 $^{w}/_{o}$. Therefore, the uranyl fluoride/water mixture assumption provides additional conservatism in this case.

5.2.1.3.4 Vessel Movement Assumption

The interaction controls placed on movement of vessels containing enriched uranium are specified in the facility procedures. In general, any item in movement (an item being either an individual vessel or a specified batch of vessels) must be maintained at 60 cm (23.6 in) edge separation from any other enriched uranium, and that only one item of each type, e.g., one trap and one pump, may be in movement at one time. These spacing restrictions are relaxed for vessels being removed from fixed positions, when one vessel may approach adjacent fixed plant without spacing restriction. The exceptions are discussed in the relevant portions of Section 5.3.

5.2.1.3.5 Pump Free Volume Assumption

There are two types of pumps used in product and dump systems of the plant:

- The vacuum pumps (product and dump) are rotary vane pumps. In the enrichment plant fixed equipment, these are assumed to have a free volume of 14 L (3.7 gal) and are modeled as a cylinder in MONK8A (SA, 2001). This adequately covers all models likely to be purchased.
- The UF_6 pumping units are a combination unit of two pumps, one 500 m³/hr (17,656 ft^3/hr) pump with a free volume of 8.52 L (2.25 gal) modeled as a cylinder, and a larger 2000 m³/hr (70,626 ft^3/hr) pump which is modeled explicitly according to manufacturer's drawings.

5.2.1.4 Nuclear Criticality Safety Analyses

Nuclear criticality safety is evaluated for the design features of the plant system or component and for the operating practices that relate to maintaining criticality safety. The evaluation of individual systems or components and their interaction with other systems or components containing enriched uranium is performed to assure the criticality safety criteria are met. The nuclear criticality safety analyses below and the safe values in Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, provide a basis for the plant design and criticality hazards identification performed as part of the integrated safety analysis described in Chapter 3, Integrated Safety Analysis Summary.

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safe values of Table 5.1-1, are applied to the facility design to prevent a nuclear criticality event. The NEF is designed and operated in accordance with the parameters provided in Table 5.1-2. The integrated safety analysis reviewed the facility design and operation and identified Items Relied On For Safety to ensure that criticality does not pose an unacceptable risk.

Where there are significant in-process accumulations of enriched uranium as UF_6 the plant design includes multiple features to minimize the possibilities for breakdown of the moderation

control limits. These features eliminate direct ingress of water to product cylinders while in process.

The calculated values of k_{eff} provided in Section 5.3 were obtained using the criticality code MONK8A (SA, 2001), in conjunction with the JEF2.2 nuclear data library. All values of k_{eff} given in Section 5.3 are equal to $k_{calc} + 3\ \sigma_{calc}$ with a safety limit of 0.95 as described in Section 5.2.1.2, Limits on Control and Controlled Parameters.

5.2.1.5 Additional Nuclear Criticality Safety Analyses

The NEF NCS analyses in Section 5.3 were performed using the above methodologies and assumptions. Any additional or future analyses will meet the following criteria:

- NCS analyses will be performed using acceptable methodologies.
- Methods will be validated and used only within demonstrated acceptable ranges.
- The analyses will adhere to ANSI/ANS-8.1-1998 (ANSI, 1998a) as it relates to methodologies.
- The intent of the validation report statement in Regulatory Guide 3.71 (NRC, 1998) will be met.
- A specific reference to (including the date and revision number) and summary description of either a manual or a documented, reviewed, and approved validation report for each methodology will be included. Any change in the reference manual or validation report will be reported to the NRC by letter.
- The reference manual and documented reviewed validation report will be kept at the facility.
- The reference manual and validation report will be incorporated into the configuration management program.
- The NCS analyses will be performed in accordance with the methods specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Section 5.4.3.4, will be used to evaluate NCS accident sequences in operations and processes.
- The acceptance criteria in NUREG-1520 (NRC, 2002), Section 3.4, as they relate to: identification of NRC accident sequences, consequences of NRC accident sequences, likelihood of NRC accident sequences, and descriptions of IROFS for NRC accident sequences will be met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety will be used.

- As stated in ANSI/ANS-8.1-1998 (ANSI, 1998a), process specifications will incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

- The following national standards, as they relate to these requirements: ANSI/ANS-8.7-1998 (ANSI, 1998b), and ANSI/ANS-8.10-1983 (ANSI, 1983b), as modified by Regulatory Guide 3.71 (NRC, 1986) will be used.

- If administrative k_{eff} margins for normal and credible abnormal conditions are used, NRC pre-approval of the administrative margins will be sought.

- Subcritical limits for k_{eff} calculations such that: k_{eff} subcritical = 1.0 - bias margin, where the margin includes adequate allowance for uncertainty in the methodology, data, and bias to assure subcriticality will be used.

- Studies to correlate the change in a value of a controlled parameter and its k_{eff} value will be performed. The studies will include changing the value of one controlled parameter and determining its effect on another controlled parameter and k_{eff}.

- An NCS program that ensures double contingency protection will be implemented. Double contingency protection will be used in determining NCS controls and IROFS. The double contingency protection will be evaluated considering the contents of ANSI/ANS-8.1-1998 (ANSI, 1998a) and the likelihood discussion contained in NUREG-1520 (NRC, 2002) Chapter 3, including consideration of the following guidance:
  - Adherence to double contingency protection: Each process that has accident sequences that could result in an inadvertent nuclear criticality shall have double contingency protection. Double contingency protection may be provided by either: (i) at least two-parameter control (the control of at least two independent process parameters) or (ii) single-parameter control (a system of multiple independent controls on a single process parameter). The first method is the preferred approach because of the difficulty of preventing common-mode failure when controlling only one parameter.
  - As used in double contingency protection, the term “concurrent” means that the effect of the first process change persists until a second change occurs, at which point the process could have an inadvertent nuclear criticality. It does not mean that the two events initiating the change must occur simultaneously. The possibility of an inadvertent nuclear criticality can be markedly reduced if failures of NCS controls are rapidly detected and the processes rendered safe. If not, processes can remain vulnerable to a second failure for extended periods of time.

  - The acceptance criteria in NUREG-1520 (NRC, 2002) Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, will be met.

5.2.1.6 Nuclear Criticality Safety Evaluations (NCSE)

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes,

operating procedures, management measures), that involves or could affect uranium, a NCSE shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with approved margin for safety) under both normal and credible abnormal conditions. If this condition cannot be shown with the NCSE, either a new or revised NCS analysis will be generated that meets the criteria, or the change will not be made.

The NCSE shall determine and explicitly identify the controlled parameters and associated limits upon which NCS depends, assuring that no single inadvertent departure from a procedure could cause an inadvertent nuclear criticality and that the safety basis of the facility will be maintained during the lifetime of the facility. The evaluation ensures that all potentially affected uranic processes are evaluated to determine the effect of the change on the safety basis of the process, including the effect on bounding process assumptions, on the reliability and availability of NCS controls, and on the NCS of connected processes.

The NCSE process involves a review of the proposed change, discussions with the subject matter experts to determine the processes which need to be considered, development of the controls necessary to meet the double contingency principle, and identification of the assumptions and equipment (e.g., physical controls and/or management measures) needed to ensure criticality safety.

Engineering judgment of the criticality safety engineer is used to ascertain the criticality impact of the proposed change. The basis for this judgment is documented with sufficient detail in the NCSE to allow the independent review by a second criticality safety engineer to confirm the conclusions of the judgment of results. Once the NCSE is completed and the independent criticality safety engineer evaluation is performed and documented, the HS&E Manager approves the NCSE. Only criticality safety engineers who have successfully met the requirements specified in the qualification procedure can perform the NCSE and associated independent review.

The above process for NCSEs is in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996).

5.3 NUCLEAR CRITICALITY SAFETY ANALYSES RESULTS

The calculated values of k_{eff} provided in this section were obtained using the criticality code MONK8A (SA, 2001), in conjunction with the JEF2.2 nuclear data library. All values of k_{eff} given in the sections below are equal to $k_{calc} + 3\ \sigma_{calc}$ with a safety limit of 0.95 as described in Section 5.2.1.2, Limits on Control and Controlled Parameters.

5.3.1 Centrifuges and Cascades

Criticality safety of TC-12 centrifuges was assessed assuming 6 $^{w}/_{o}$ ^{235}U enrichment. The only potential for a criticality incident in a centrifuge cascade is by gross uranium accumulation in failed centrifuges. To achieve criticality in a cascade would require an array of failed centrifuges to be completely filled with uranic breakdown (as $UO_2F_2 \cdot 3.5H_2O$). The extreme conditions required to obtain the necessary uranic accumulation for criticality by this mechanism could never credibly occur in practice. Furthermore, it is highly unlikely that: (1) the centrifuges in such an array would fail simultaneously, (2) the failures would lead to inleakage of moist air into the failed centrifuges, (3) all the failed centrifuges would fill up with UF_6 breakdown products, and (4) would have an H/U ratio that is near optimum. Therefore, the possibility of a criticality incident in a centrifuge cascade can be considered not credible.

5.3.2 Product Cylinders

The product enrichment within a 48Y or 30B product cylinder is limited to 5.0 $^{w}/_{o}$ ^{235}U by the plant design, configuration and operating features. The UF_6 content is limited to no more than the 48Y or 30B cylinder fill limit by the plant design and operating features. The moderation within the cylinder is controlled by a series of plant operating features. These features include, among others, checks that the cylinder is clean and empty prior to the commencement of fill. Also, the moderator (H_2O, HF) entering the cylinder is monitored during the time the cylinder is connected to the plant UF_6 systems.

Calculations were performed on infinite two-dimensional arrays of full 48Y or 30B product cylinders. Inside each cylinder a region of UO_2F_2/water mixture was located. The remainder of the interior of the cylinder was assumed to be filled with 6.0 $^{w}/_{o}$ ^{235}U enriched UF_6. Cylinders in the arrays were placed with the valve and base ends alternately in contact, so that the moderated region in a given cylinder was in the closest possible proximity to the moderated region in an adjacent cylinder. All cylinders were considered to be lying on a concrete pad one meter thick. Moderation was varied to obtain the optimum H/U ratio. Worst-case external reflection/moderation conditions were found by varying the density of the interstitial water between cylinders to simulate frost or snow. The calculation also assumed one cylinder above (touching) the array to simulate movement in/out/over the array.

For the 48Y cylinder, the condition that met the upper safety limit had an H/U ratio of 11.5 with an interstitial water density of 0.10 g/cm^3 (6.2 lb/ft^3). Thus, the maximum safe mass of hydrogen in each type product 48Y cylinder in an array was determined to be 1.05 kg (2.31 lb) present in the form of 9.5 kg (20.9 lb) of water.

For the 30B cylinder, the condition that met the upper safety limit had an H/U ratio of 10.5 with an interstitial water density of 0.25 g/cm^3 (15.6 lb/ft^3). Thus, the maximum safe mass of

hydrogen in each type product 30B cylinder in an array was determined to be 0.95 kg (2.09 lb) present in the form of 8.5 kg (18.7 lb) of water.

Criticality safety of Type 48Y and 30B product cylinders depends on the control of moderator content. Criticality safety is achieved by ensuring that there is less than 1.05 kg (2.31 lb) of hydrogen present in a Type 48Y cylinder and less than 0.95 kg (2.09 lb) of hydrogen present in a Type 30B cylinder (as shown in Table 5.1-2, Safety Criteria for Buildings/Systems/ Components).

5.3.3 Product Vent Subsystem UF_6 Cold Traps

During venting of product cylinders any UF_6 not condensed out in the product take-off cylinder will be removed from the vent gas stream by the Product Vent Subsystem. The majority of the UF_6 will condense out in a cold trap. The body is encased by insulating material. Although the traps have a large internal volume they are individually safe by shape, the trap body having an internal diameter of 20.3 cm (8.0 in). This compares with the safe diameter of 21.9 cm (8.6 in) for 6.0 $^{w}/_{o}$ enrichment (as shown in Table 5.1-2). Individual cold traps are thus safe in isolation for any uranyl fluoride/water mixture. In practice the maximum H/U atom ratio in the cold traps will be 7; however, a sensitivity study is performed to determine the optimum H/U ratio, providing an additional margin of safety.

The cold trap and the standby cold trap are separated from each other by center-to-center separation of 110 cm (43.3 in). There is a minimum edge separation of 180 cm (70.9 in) from any other fixed plant vessels that can accumulate enriched uranium. The pair of traps can thus be considered to be neutronically isolated from other fixed vessels.

Calculations were performed on the isolated pair of cold traps and were found to be substantially subcritical with k_{eff} = 0.8030. The calculations assumed an enrichment of 6.0 $^{w}/_{o}$, H/U of 7 and 2.5 cm (1.0 in) water reflection placed at the model boundary to simulate spurious reflection.

According to the restrictions on movement of mobile vessels discussed in Section 5.2.1.3.4, Vessel Movement Assumption, one vessel can come into contact with a trap but any others have to be kept at 60 cm (23.6 in) separation.

MONK8A (SA, 2001) calculations have been performed in which a vacuum cleaner is in contact with one of the product cold traps, and another vessel (a 14 L (3.7 gal) product vent vacuum pump) is at 60 cm (23.6 in) edge spacing from the same cold trap. These are typical of Separation Plant mobile vessels. Each mobile vessel was modeled with the appropriate uranic fill; the vacuum cleaner was filled with uranyl fluoride/water mixture with optimum moderation (H/U=12), and the vacuum pump (conservatively containing hydrocarbon oil) was filled with uranic breakdown of composition $UF_4 \cdot 10.5CH_2$. The resulting k_{eff} = 0.8229 shows a slight increase in reactivity with respect to the isolated pair of traps using the same conservative assumptions.

Additionally, calculations were performed in which it was assumed that there are no movement controls, and both the vacuum cleaner and pump were in contact with one of the cold traps. Even with 2.5 cm (1.0 in) spurious water reflection placed around each unit, and at enrichment of 6.0 $^{w}/_{o}$, the result remained substantially subcritical with k_{eff} = 0.8673.

The product cold traps have therefore been determined to be safe both as a pair in isolation and while interacting with other fixed plant or vessels in movement for ^{235}U enrichments up to 6.0 $^{w}/_{o}$.

5.3.4 Product Vent Subsystem Pumping System

The pumping system associated with each product cold trap consists of one vacuum pump, two chemical absorber traps for UF_6 and two oil traps. Each pump and set of traps is positioned on an individual frame (skid). On the inlet side of the vacuum pump is a 10 kg (22.0 lb) (carbon) capacity activated carbon vent trap. The main function of the vent trap is to remove UF_6 from the gas stream. Gas from the carbon trap then passes through a similarly sized alumina vent trap to remove any remaining HF. Between the two traps and the pump is a smaller oil trap. A small oil trap is also fitted to the exhaust side of the pump. These product vent traps are individually safe by diameter (20.3 cm (8.0 in) compared with the safe diameter of 21.9 cm (8.6 in) calculated for 6.0 $^{w}/_{o}$ enrichment) (as shown in Table 5.1-2). However, calculations have been performed concerning the effect of possible neutron interaction with nearby (uranium bearing) equipment.

In the MONK8A (SA, 2001) calculations for the Product Vent Subsystem the plant spacing to the edge of the standby vent system is assumed to be 50 cm (19.7 in). The standby vent system has been included in the model. The traps were both assumed to fill entirely with uranyl fluoride/water with no restriction on water content. This is conservative, as in practice the H/U ratio of the uranyl fluoride in the traps will have a limiting upper value of 7 as described in Section 5.2.1.3.3, Uranium Accumulation and Moderation Assumption. Also, the space within the trap, which would normally be occupied by carbon or alumina, is modeled as being filled with uranic material. This maximizes the mass of fissile material within the traps and provides added conservatism. The pump, alumina traps, oil trap and exhaust filter are assumed to be filled with uranyl fluoride/water of unlimited water content. This is conservative, as virtually no uranium is expected in these components.

Calculations were performed to account for interaction with other vessels in movement. According to the restrictions on movement, one mobile vessel can come into contact with one of the fixed chemical absorber traps, but other mobile vessels are assumed to be at 60 cm (23.6 in) separation. The case modeled was for a vacuum cleaner (of diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in)) to be brought into contact with the vacuum pump in the product vent array. One other item, a 14 L (3.7 gal) rotary vane pump, was placed at 60 cm (23.6 in) edge spacing from the cleaner.

The MONK8A (SA, 2001) calculation for the worst case, where all vessels were assumed to be entirely filled with uranyl fluoride/water mixture at optimum moderation, a trap and a vacuum cleaner are in contact with one of the fixed pumps, and all pumps were modeled with volumes of 14 L (3.7 gal), yields a k_{eff} = 0.9328.

It should be noted that the above MONK8A (SA, 2001) model represents extreme accident conditions in terms of uranium accumulation and moderator ingress. It should also be noted that the simple MONK8A (SA, 2001) model used for the vacuum pump in all of the calculations is conservative. Since the real shape of the internal free volume is far from optimum, an explicit model of the pump is expected to result in a significant reduction in k_{eff}.

The product vent pumping system has been shown to be safe under normal operating conditions and credible abnormal operating conditions, for ^{235}U enrichments up to 6.0 $^{w}/_{o}$.

5.3.5 Contingency Dump Trap

Each cascade of centrifuges is connected to an independent Contingency Dump System. Included in the Contingency Dump System are three contingency dump traps for each cascade.

The average enrichment of the UF_6 being dumped from a cascade depends on the product and tails enrichments. Within the ranges of product enrichment up to 5.0 $^{w}/_{o}$ ^{235}U and tails depletion to 0.34 $^{w}/_{o}$ ^{235}U the average enrichment of the UF_6 being dumped is always less than 1.5 $^{w}/_{o}$ ^{235}U. Based on this, the contingency dump traps will be analyzed at an enrichment of 1.5 $^{w}/_{o}$ rather than 6.0 $^{w}/_{o}$. The contingency dump traps are sodium fluoride traps with an inside diameter of approximately 54 cm (21.3 in).

MONK8A (SA, 2001) calculations have been carried out first for an isolated trap with 2.5 cm (1.0 in) of water reflection around the trap body. The model assumed that adsorbed UF_6 within the trap is converted to $UO_2F_2{\cdot}3.5H_2O$, i.e., the accident condition with air inleakage. The uranium enrichment was 1.5 $^{w}/_{o}$ ^{235}U. The value of k_{eff} obtained was 0.6466. The model represents a UF_6 loading in the trap of approximately 220 kg (485 lb), which would require many dumps to achieve. Contingency dump traps are thus intrinsically safe by a very large margin.

Considering interaction between the three closely spaced traps, criticality safety is demonstrated by comparison with the MONK8A (SA, 2001) calculations for storage of contingency dump traps in unspaced linear arrays. The calculation modeled a linear array of seven touching dump traps with three other vessels at 60 cm (23.6 in) spacing from the array (a residue container, a vacuum cleaner cylinder and a UF_6 pump unit). An additional dump trap was also placed in contact with the center trap of the linear array. The value of k_{eff} obtained was 0.8537. The modeled arrangement is more conservative than three spaced traps interacting with the same mobile vessels and it can be concluded that contingency dump traps are safe when interacting with any mobile vessels that are likely to be present.

5.3.6 Product Pumping Train UF_6 Pumps

Low pressure UF_6 pumping is carried out by combination pumping units, the Product Pumping Trains. More than 200 cm (78.7 in) separates each Product Pumping Train in the plant from other uranium containing vessels, so only interaction with mobile components needs be considered. Additionally, when being removed for repair or maintenance, a UF_6 pump might pass near to another similar pump.

The currently planned pump combination unit consists of two Leybold pumps, models WS2000 series and WS500 series, positioned in a fixed frame. The WS500 series has an internal free volume of 8.52 L (2.25 gal), which is less than half of the maximum safe volume of 18 L (4.8 gal) at 6.0 $^{w}/_{o}$ enrichment (as shown in Table 5.1-2). Therefore the WS500 series pump can be modeled conservatively as an isometric cylinder of the same volume. However, the WS2000 series pump has an internal free volume of 33 L (8.7 gal), which considerably exceeds the safe volume (as shown in Table 5.1-2), and even exceeds the minimum critical volume of 24 L (6.3 gal) (as shown in Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2). Although the WS2000 series pump has a larger than critical internal free volume, the shape of the internal volume is far from the optimum. Therefore, the WS2000 pump was modeled in some detail based on drawings supplied by the manufacturer.

MONK8A (SA, 2001) calculations were initially performed for an isolated pump combination to assess the intrinsic safety of the combination. The maximum k_{eff} of 0.7479 was achieved using an enrichment of 6.0 $^{w}/_{o}$ and an optimum H/U ratio of 12. From this analysis, the pump combination in isolation can be regarded as being intrinsically safe. As mentioned above, there is potential for a second pump unit to approach when being removed for maintenance. Calculations were performed on pairs of pumps in contact with each other, either side by side, or touching at the gearbox ends. The most reactive case was with the gearbox ends touching (k_{eff} = 0.8277), assuming an enrichment of 6.0 $^{w}/_{o}$ and an optimum H/U ratio of 10.

To consider interaction of mobile vessels, calculations were performed which added a vacuum cleaner to the pair of pumps, either in contact with the gearbox end (with the pumps side by side) or alongside one of the pumps (with the pumps touching at the gearbox ends). The worst case was achieved with the latter arrangement giving a k_{eff} = 0.8444.

A 14 L (3.7 gal) isometric cylinder representing an additional pump in transit was then placed 60 cm (23.6 in) from the vacuum cleaner resulting in a k_{eff} = 0.8743. This increase reflects the fact that the 14 L (3.7 gal) pump is the most reactive unit in the array; over 80% of fission events occur inside the 14 L (3.7 gal) pump. The relative orientation of the product pumps and vacuum cleaner has little effect on the value of k_{eff} when the 14 L (3.7 gal) pump is present.

Even assuming the most conservative geometry and moderation conditions, k_{eff} remains substantially subcritical. Note that the movement of vessels considered above is considered to be part of normal operating conditions. The abnormal operating condition pertaining to the vessels concerns the assumption that all the vessels are completely filled with uranic breakdown at optimum moderation. This would be extremely unlikely for a single vessel in the array, and even more unlikely for more than one vessel.

It can be concluded that:

- An array of two pump units is safe at any spacing. No restriction is placed on the moderator content of the pump units.
- One pump or pump unit may be moved, and may approach another similar pump unit or vacuum cleaner (of safe diameter) at any orientation, and without spacing restrictions. Other pumps (of 14 L (3.7 gal) internal volume or less) must not approach within 60 cm (23.6 in) of a product pumping train. No restriction is placed on the moderator content of any of the vessels.

5.3.7 Vacuum Cleaners

Subject to management authorization, vacuum cleaners will be used throughout the plant for general cleaning and for removal of the small amounts of breakdown that might arise when components are disconnected from plant.

Within the individual nuclear criticality evaluations for plant items, the MONK8A (SA, 2001) calculations have included a vacuum cleaner next to plant items. The vacuum cleaner was assumed to be a cleaner of internal diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in) and was assumed to be entirely filled with uranic material with an enrichment of 6.0 $^{w}/_{o}$. MONK8A (SA, 2001) calculations have been carried out for an isolated cylinder using these dimensions, filled with uranyl fluoride/water at optimum moderation and with 2.5 cm (1.0 in) water reflection. This gave a value for k_{eff} of 0.8037. The cleaner has high efficiency particulate air (HEPA)

filtration on the exhaust, and will be dedicated for cleaning operations where uranic material is involved and will be marked clearly.

5.3.8 Technical Services Building Solid Waste Collection Room

Operation of the plant generates solid low level radioactive waste, principally spent chemical absorbers and contaminated combustible wastes such as tissues, gloves and overalls from maintenance operations. These wastes are stored in drums and pots in the TSB waste storage room.

Product traps with 10 kg (22.0 lb) of activated carbon will be discharged in the Ventilated Room into 12 L (3.2 gal) containers, which will then be stored in an engineered array in the solid waste collection room.

MONK8A (SA, 2001) calculations were performed for a single 12 L (3.2 gal) residue container holding charcoal/uranyl fluoride/water mixture over a range of H/U ratios. The container was modeled as an equiaxed cylinder of radius 12.4 cm (4.9 in) and height 24.8 cm (9.8 in) which was placed on a 20 cm (7.9 in) thick concrete layer with reflection beneath the lower face to simulate infinite depth of concrete. The cylinder volume was completely filled with the charcoal/uranic mixture. A 2.5 cm (1.0 in) thick water layer enclosed the cylinder sides and top surface. At the optimum H/U ratio of 24, the value of k_{eff} is 0.7025 compared with a maximum value for k_{eff} of 0.8570 for an isolated 12 L (3.2 gal) cylinder of oil/UF_4 mixture. This indicates that the charcoal mixture will be safe when stored in 12 L (3.2 gal) containers.

For the array, a 5x5 horizontal array of cylinders was modeled explicitly with an additional container in contact with the center cylinder of the 5x5 unit to simulate accidental movement of an extra container into a storage array. The containers were modeled resting on a 30 cm (11.8 in) thick concrete layer and a 2.5 cm (1.0 in) water reflector was placed around each container. The uranic/oil mix was at an H/U ratio of 21. The value of k_{eff} obtained for the array model was 0.9281.

Therefore, arrays of up to 5x5 12 L (3.2 gal) containers containing chemical absorber material are therefore safe under worst-case conditions with 60 cm (23.6 in) spacing between containers.

5.3.9 Technical Services Building Ventilated Room

The purpose of the TSB Ventilated Room is for the emptying of chemical traps and to deal with faults associated with cylinder valves. Valves may be removed from cylinders and new valves fitted. The replaced valve is then tested for vacuum integrity.

Calculations have been performed on storage arrays of product vent chemical (carbon) traps. The calculations also cover the storage of alumina traps, which are of similar dimensions but have a lower uptake of uranium. The alumina traps are not normally exposed to uranium (their purpose is to remove HF), but it is possible that an alumina trap could be connected to the plant by mistake in place of a carbon trap. The modeling of alumina traps as carbon traps covers this possibility.

The chemical traps are essentially empty steel cylinders into which steel internal parts including perforated plates to carry the activated carbon are placed. The activated carbon in the trap

adsorbs UF_6 which, is then assumed to be hydrated (by moist air inleakage) to form a uranyl fluoride/water mixture with a maximum H/U ratio of 7.

The traps are of internal diameter 20.3 cm (8.0 in) and height 105.8 cm (41.7 in). The diameter is less than the maximum safe diameter (21.9 cm (8.6 in)) for 6.0 $^{w}/_{o}$ enriched material (as shown in Table 5.1-2) for a single cylinder. However, it is possible that large numbers of traps (e.g. in storage arrays) are more reactive.

Arrays of chemical traps were modeled using MONK8A (SA, 2001). An array of 7x7 traps and a vacuum cleaner yields a k_{eff} of 0.9191 assuming 6.0 $^{w}/_{o}$ enrichment. This was modeled with the sidewalls of the traps touching, which could not happen in practice since there is a lip at the top. Taking account of one of these lips to give 5 cm (2.0 in) spacing between the traps an array of 11x11 traps and a vacuum cleaner was modeled and gave a reduced k_{eff} of 0.8665.

It can be concluded that arrays of these chemical traps containing uranium of up to 6.0 $^{w}/_{o}$ enrichment are safe up to an11x11 array configuration with no spacing restriction. The only stipulation that needs to be made is that stacking of traps in an array is not allowed.

5.3.10 Technical Services Building Chemical Laboratory

The prime function of the TSB Chemical Laboratory is to analyze the product material to ensure that it meets the product purity specification. This involves the handling and storage of a large number of 1S sample bottles and the production of hydrolyzed UF_6 solutions for the subsequent analysis. There may also be a requirement for this laboratory to deal with other samples, for example, those from the Decontamination System's tanks. These samples will have uranium concentrations much less than the hydrolyzed UF_6 solutions considered below and as such can be treated in the same manner. There may be a requirement for other solid samples to be analyzed such as deposits removed from plant components prior to decontamination and these can be dealt with on a formal mass accountancy basis. The double batching mass limit of 45% of the minimum critical mass is used in the nuclear criticality safety for these samples.

Samples of UF_6 are typically received in 1S cylinder sample bottles. The storage system for 1S bottles is a rack system within two storage areas of approximate dimensions of 1 meter wide and 2.5 m (8.2 ft) high. These have a combined total of 168 slots and normally up to three bottles would be placed in one slot. The normal capacity is approximately 500 bottles.

The 1S sample bottles containing enriched uranium are filled from 30B product cylinders that are subject to criticality control on moderator content such that the H/U ratio of the contents does not exceed unity. Product UF_6 is controlled to a product specification of 99.5% UF_6 purity equivalent to an H/U ratio of only 0.088 (impurity assumed to be all HF). Although 1S bottles are not strictly moderator controlled in the sense that product cylinders are, their contents are expected to be representative of the UF_6 product.

A full 30B cylinder contains more than 2,000 kg (4,409 lb) of UF_6 that is safe at an H/U ratio of unity. The maximum permitted fill for 1S bottles is 450 g (0.99 lb) UF_6 (plant fill limit is 400 g (0.88 lb)) and therefore approximately 4,000 filled 1S bottles are equivalent to one full 30B cylinder.

The storage array can be considered as a heterogeneous system of vessels in a regular lattice arrangement. A series of MONK8A (SA, 2001) calculations have been carried out for generic storage of 1S bottles which include lattice arrangements of 1S bottles in array sizes exceeding

the number stored in the laboratories. These calculations modeled a 25x25 array of 1S bottles containing 450 g (0.99 lb) of $UF_6 \cdot HF$ mixture at 6.0 $^{w}/_{o}$ ^{235}U enrichment. The array was modeled at optimum spacing (triangular pitch) and the water reflection conditions considered included water flooding although flooding of the array is not credible. The most reactive case from the above MONK8A (SA, 2001) calculations gave a value for k_{eff} of 0.6549.

Based on this analysis, the 1S sample bottles are safe in an array.

5.3.11 Technical Services Building Decontamination Workshop

The Decontamination Workshop will essentially be used for the dry stripping of pumps. Entry to the facility is via an airlock door leading into the main stripping room. The stripping room comprises an overhead crane for lifting large pumps and pump sets, two hydraulic worktables on which pumps will be disassembled and cleaned, and an engineered location for storage containers. The degassing cubicle is ventilated by the Gaseous Effluent Vent System (GEVS) system, a ventilation system containing HEPA filtration.

A MONK8A (SA, 2001) calculation was performed for a generic model of arrays of pumps of volume not exceeding 14 L (3.7 gal). The pumps were modeled simply as equiaxed cylinders of uranic material of 14 L (3.7 gal) volume surrounded by an iron annulus of 0.5 cm (0.2 in) thickness representing typical casing thickness for UF_6 pumps. The uranic material was uranyl fluoride/water mixture at an H/U atomic ratio of 7 and at 6.0 $^{w}/_{o}$ ^{235}U enrichment. The two pumps were modeled in contact along the cylinder wall and a 2.5 cm (1.0 in) water annulus was included around each pump to simulate spurious reflection. The pumps were positioned above a 20 cm (7.9 in) thick concrete layer but separated from this layer by the 2.5 cm (1.0 in) water thickness. Reflection was used to simulate an infinite linear array of pump pairs with 60 cm (23.6 in) edge spacing between pairs. The resulting value of k_{eff} was 0.8552.

The MONK8A (SA, 2001) model described above was modified to replace the two touching 14 L (3.7 gal) pumps with two pump sets in contact, each pump set assumed a combination of a Leybold WS2000 pump and a Leybold WS500 pump.

For the WS2000 pump, a detailed model was used based on the actual construction of the pump from information supplied by the manufacturer. Some conservative modeling simplifications were required mainly for the gearbox and motor assemblies. In the case of the WS500 pump, which has a free volume of 8.52 L (2.25 gal), the simple equiaxed cylinder model was used. To simulate spurious reflection, the two pumps were modeled as being separated vertically by the thickness of the 2.5 cm (1.0 in) water annulus around the cylinder of the WS500 pump. This spurious reflection assumption is less than the actual separation when the pumps are in their pump frame.

Two pump sets were explicitly modeled but rather than being side by side as was the case for the 14 L (3.7 gal) pumps, the pump sets were orientated in line such that the WS2000 pumps were in contact at the gearbox end. Sensitivity studies show this to be the most reactive configuration for the pump sets. It was also assumed that the pumps were filled with the same uranic mixture as for the 14 L (3.7 gal) pumps and the infinite linear array was again used. The resulting value of k_{eff} was 0.8202.

Therefore, a linear array of pump pairs with 60 cm (23.6 in) spacing is safe.

5.3.12 Technical Services Building Fomblin Oil Recovery System

The Fomblin Oil Recovery System is used to treat the drained oil from the process pumps. The system is operated as a batch process, one batch being up to 12 L (3.2 gal) of oil. Only one batch of oil is processed at any one time representing a maximum of 12 L (3.2 gal). The maximum volume of any vessel on the rig is again 12 L (3.2 gal) and is thus intrinsically safe. However, MONK8A (SA, 2001) calculations demonstrate that the rig would remain safe even if all vessels were completely filled with uranyl fluoride-water mixture at 6.0 $^{w}/_{o}$ enrichment and at optimum moderation. Uranyl fluoride/water mixture is more conservative than a Fomblin oil/UF_4 mixture. In the Fomblin oil/UF_4 mixture, dissolved HF provides the moderation and HF solubility in Fomblin oil is extremely low.

The MONK8A (SA, 2001) calculations for the rig modeled the fixed vessels in their normal positions and included one 12 L (3.2 gal) container adjacent to the first mixing vessel to represent the batch of oil being moved to the rig. A 2.5 cm (1.0 in) water layer was modeled around the vessels to simulate spurious reflection. All vessels contained uranyl fluoride-water mixture as stated above, and a range of H/U atomic ratios were considered to determine the optimum moderation. The maximum value of k_{eff} for the calculations was 0.7976 at an H/U ratio of 14.

5.3.13 Separations Building Gaseous Effluent Vent System

The function of the Separations Building Gaseous Effluent Vent System (GEVS) is to remove HF and trace quantities of oil and uranic particulates from the effluent gas.

There are two sources of material to the GEVS, flexible exhaust hoses and rotary pump exhausts.

The rotary pump exhaust gas arising from the Product Vent Subsystem passes from the UF_6 cold trap through the activated carbon trap and alumina trap and finally through the rotary pump.

Excessive carry over from the cold trap to the carbon trap is avoided by the closure of a valve in the interconnection by a low pressure or a high temperature trip in the cold trap. The exhaust gas then passes through a trap filled with carbon that reacts irreversibly with the UF_6 and then passes through an activated alumina to remove HF. The gas is then pumped out into the GEVS for final clean up. These chemical traps are replaced at regular intervals or when the weight indicators show that there is significant build up of material. A weight trip on the carbon trap isolates the process line from the GEVS when the traps are about to become saturated.

The flexible exhaust hoses will be used to support product (and feed and tails) cylinder and pump changeout and maintenance activities in the separations plant and trace enriched particulate matter may be released.

The potentially oil bearing inflow to the GEVS from the rotary vacuum pumps exhausts is first passed through an electrostatic precipitator to remove the aerosol oil before joining the rest of the effluent gas. It then passes through pre filters, HEPA filters for particulates removal and impregnated carbon filters for removing HF. Prior to the HEPA filters there is a fluoride monitor that will alarm if the concentration of the fluorine compounds within the air being drawn into the filters exceeds a pre-determined level. This will provide assurance that accumulation of uranium in the filters is not occurring. The filters are equipped with differential pressure indicators and ^{235}U selective gamma monitors that will trip on blockage or build-up of material.

The amount of uranium in the electrostatic precipitator will also be monitored for gamma radiation to ensure that any slow, chronic accumulation of fissile material does not pose a hazard. The gamma monitors are deemed IROFS.

The carbon trap weight trip and GEVS filter gamma detector are installed to prevent any potential for criticality. In addition, the accumulation rate of uranium in the GEVS is very low compared with the safe mass of 12.2 kg U (26.9 lb U) assuming double batching and all the uranium were enriched to 6.0 $^{w}/_{o}$. These low accumulations coupled with the weight trip and gamma detectors render a criticality accident in the GEVS highly unlikely.

5.3.14 Technical Services Building Gaseous Effluent Vent System

The GEVS for the Technical Services Building (TSB) is designed to remove particulate material, mainly UO_2F_2, and hydrogen fluoride from the various operations within the building. Air from these operations is drawn into the GEVS and passes through HEPA filters and a carbon filter impregnated with potassium carbonate. Only one bank of filters exists and if there are high levels of emissions of HF or uranium, then operations in the TSB are terminated. A hydrogen fluoride detector is located both before and after the filters; a ^{235}U selective gamma monitor is located at the filters. This gamma monitor trips the TSB GEVS and is deemed an IROFS.

Within the TSB Ventilated Room, chemical traps will be emptied and product cylinders may be brought into the room for valve changes and subsequent testing. In the case of the traps there will be a mixture of product, feed and dump traps with a few from the tails operations. The product traps will be 10 kg (22.0 lb) carbon traps with a maximum holdup of 12 kg (26.5 lb) UF_6. The traps will have been de-gassed prior to being removed from the plant and there will be very little of the UF_6 absorbed on the trap that could become airborne. There may be a small amount of carbon drawn into the system as a result of emptying the traps. With approximately 20 carbon traps processed per year it is not considered credible that kilogram quantities of uranium would be drawn into the GEVS, before filters were changed out.

A possible scenario for the acute accumulation of enriched uranium from the Ventilated Room exists from the valve testing operations. For this operation a cylinder is taken into the room and the valve is removed. A new valve is fitted to the cylinder and the cylinder is then pressure tested. This involves pressurizing the container with nitrogen then evacuating. For this operation the cylinder is connected to a portable rig, which in turn exhausts to the GEVS. Since all pumps are lubricated with a UF_6 compatible oil there is the remote possibility that UF_6 could be pumped directly from the cylinder to the GEVS. Weight and temperature trips on the carbon trap in this rig prevent this transfer from occurring. These weight and temperature trips are deemed IROFS.

Within the TSB Decontamination System there are a number of cleaning tanks. Components entering these tanks will have either been cleaned or de-gassed. It is not considered likely that significant quantities of uranium would enter the GEVS as a result of these decontamination operations or the subsequent processing of the residues. The facility also provides the plant with a sample bottle cleaning service. Type 1S sample bottles delivered to the facility will be cleaned provided that there is no more than 20 g (0.04 lb) of residual material within the bottles. Even if this was all UF_6 and the bottle was opened the operator would see white hydrogen fluoride fume and there may be some small quantity of UF_6 associated with the release. Many mal-operations would be required for the GEVS to see the quantity of material that would be needed to initiate a criticality.

Before pumps enter the TSB Contaminated Workshop there is a requirement for them to be de-gassed prior to transfer. It would be unusual for pumps to enter the facility with significant quantities of UF_6 remaining within the pump, including UF_6 dissolved in the Fomblin oil. On entering the facility the pumps are taken to the outgas area where the oil is removed. If dissolved UF_6 were present in the oil then there would be some fuming this would mainly be as a result of the dissolution of the UF_6 from the oil reacting with the water in the air. This would produce UO_2F_2 and HF. The HF would be drawn into the GEVS and the majority of the UO_2F_2 would remain with the oil. The number of product pumps that cannot be successfully de-gassed is small and it is not considered that a significant fraction of the uranium in the oil would enter the GEVS. Once the pumps have been transferred to the hydraulic table there will be uranium associated with the residual oil in the pump and some in the form of dry breakdown products. It is not considered possible that significant quantities of these will become airborne during the cleaning operations.

For the activities in the TSB, the accumulation rate of uranium in the GEVS is very low compared with the safe mass of 12.2 kg U (26.5 lb U) assuming double batching and all the uranium were enriched to 6.0 $^{w}/_{o}$. These low accumulations coupled with regular sampling of filters and the engineered safeguards render a criticality accident highly unlikely.

5.3.15 UF_6 Product Pipework

Product pipework in the Separations Building varies in size up to a maximum nominal diameter of 150 mm (5.9 in). Only minimal surface deposition of UF_6 occurs in pipework but criticality safety has been assessed for the possibility of localized blockages in pipes with the formation of uranyl fluoride due to air inleakage.

MONK8A (SA, 2001) calculations have been performed for generic arrays of pipe intersections, filled entirely with uranyl fluoride / water mixture at optimum moderation at 6.0 $^{w}/_{o}$ enrichment. The minimum permitted free space between intersections was determined to be 520 mm (20.5 in) for 150 mm (5.9 in) nominal pipe, and 135 mm (5.3 in) for 100 mm (3.9 in) nominal pipe; no spacing restriction applies to pipework of nominal diameter 65 mm (2.6 in).

The above restrictions apply to individual pipe runs with up to 64 intersections or adjacent pipe runs totaling up to 64 intersections.

Parallel pipe runs containing product material will either fit within the safe cylinder diameter of Table 5.1-1 or be explicitly modeled.

The Separations Building pipework conforms to the above specifications. If not, explicit calculations will be performed. For example, the spacing restriction might not be satisfied, but the pipework might have fewer than 64 intersections.

5.4 CRITICALITY ACCIDENT ALARM SYSTEM (CAAS)

The facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, (CFR, 2003d). Areas where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2003d) mass limits are provided with CAAS coverage. Additional information regarding the CAAS may be found in Section 3.1.5. Emergency management measures are covered in the facility Emergency Plan.

5.5 REPORTING

The following are NCS Program commitments related to event reporting:

- A program for evaluating the criticality significance of NCS events will be provided and an apparatus will be in place for making the required notification to the NRC Operations Center. Qualified individuals will make the determination of significance of NCS events. The determination of loss or degradation of double contingency protection or IROFS will be made against the license and 10 CFR 70 Appendix A (CFR, 2003f).

- The reporting criteria of 10 CFR 70 Appendix A and the report content requirements of 10 CFR 70.50 (CFR, 2003g) will be incorporated into the facility emergency procedures.

- The necessary report based on whether the IROFS credited were lost, irrespective of whether the safety limits of the associated parameters were actually exceeded will be issued.

- If it cannot be ascertained within one hour of whether the criteria of 10 CFR 70 Appendix A (CFR, 2003f) Paragraph (a) or (b) apply, the event will be treated as a one-hour reportable event.

5.6 REFERENCES

ANSI, 1998a. Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1998, American National Standards Institute/American Nuclear Society, 1998.

ANSI, 1998b. Guide for Nuclear Criticality Safety in the Storage of Fissile Materials, ANSI/ANS-8.7-1998, American National Standards Institute/American Nuclear Society, 1998.

ANSI, 1983. Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement, ANSI/ANS-8.10-1983, American National Standards Institute/American Nuclear Society, 1983.

ANSI, 1991. Nuclear Criticality Safety Training, ANSI/ANS-8.20-1991, American National Standards Institute/American Nuclear Society, 1991.

ANSI, 1997. Nuclear Criticality Safety Based on Limiting and Controlling Moderators, ANSI/ANS-8.22-1997, American National Standards Institute/American Nuclear Society, 1997.

AREVA, 2004. MONK8A Validation and Verification, AREVA, April 16, 2004.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.24, Criticality accident requirements, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.72, Facility changes and change process, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Part 70, Appendix A, Reportable Safety Events, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Section 70.50, Reporting requirements, 2003.

Lychev, 1990. Crystalline Hydrates of Uranyl Fluoride at 20°C, Lychev, Mikhalev and Suglobov. Journal of Soviet Radiochemistry, Vol 32, 1990.

NEA, 2002. International Handbook of Evaluated Criticality Safety Benchmark Experiments, NEA/NSC/DOC(95)03, Nuclear Energy Agency, September 2002 Edition.

NRC, 1998. Nuclear Criticality Safety Standards for Fuels and Materials Facilities, Regulatory Guide 3.71, U.S. Nuclear Regulatory Commission, August 1998.

NRC, 2001. Guide for Validation of Nuclear Criticality Safety Calculational Methodology, NUREG/CR-6698, U.S. Nuclear Regulatory Commission, January 2001.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

NSE, 1962. The Measurement of Σta and Other Nuclear Properties of ^{233}U and ^{235}U in Critical Aqueous Solutions, R. Gwin and D. W. Magnuson, Nuclear Science and Engineering, Volume 12, 1962.

SA, 2001. Serco Assurance, ANSWERS Software Service, "Users Guide for Version 8 ANSWERS/MONK(98) 6," 1987-2001.

TABLES

Table 5.1-1 Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2

Page 1 of 1

Parameter	Critical Value k_{eff} = 1.0	Safe Value k_{eff} = 0.95	Safety Factor
Values for 5.0 $^{w}/_{o}$ enrichment			
Volume	28.9 L (7.6 gal)	21.6 L (5.7 gal)	0.75
Cylinder Diameter	26.2 cm(10.3 in)	23.6 cm (9.3 in)	0.90
Slab Thickness	12.6 cm (5.0 in)	10.7 cm (4.2 in)	0.85
Water Mass	17.3 kg H_2O (38.1 lb H_2O)	12.7 kg H_2O (28.0 lb H_2O)	0.73
Areal Density	11.9 g/cm^2 (24.4 lb/ft^2)	9.8 g/cm^2 (20.1 lb/ft^2)	0.82
Uranium Mass	37 kg U (81.6 lb U)		
- no double batching		26.6 kg U (58.6 lb U)	0.72
- double batching		16.6 kg U (36.6 lb U)	0.45
Values for 6.0 $^{w}/_{o}$ enrichment			
Volume	24 L (6.3 gal)	18 L (4.8 gal)	0.75
Cylinder Diameter	24.4 cm (9.6 in)	21.9 cm (8.6 in)	0.90
Slab Thickness	11.5 cm (4.5 in)	9.9 cm (3.9 in)	0.86
Water Mass	15.4 kg H_2O (34.0 lb H_2O)	11.5 kg H_2O (25.4 lb H_2O)	0.75
Areal Density	9.5 g/cm^2 (19.5 lb/ft^2)	7.5 g/cm^2 (15.4 lb/ft^2)	0.79
Uranium Mass	27 kg U (59.5 lb U)		
- no double batching		19.5 kg U (43.0 lb U)	0.72
- double batching		12.2 kg U (26.9 lb U)	0.45

Table 5.1-2 Safety Criteria for Buildings/Systems/Components

Page 1 of 1

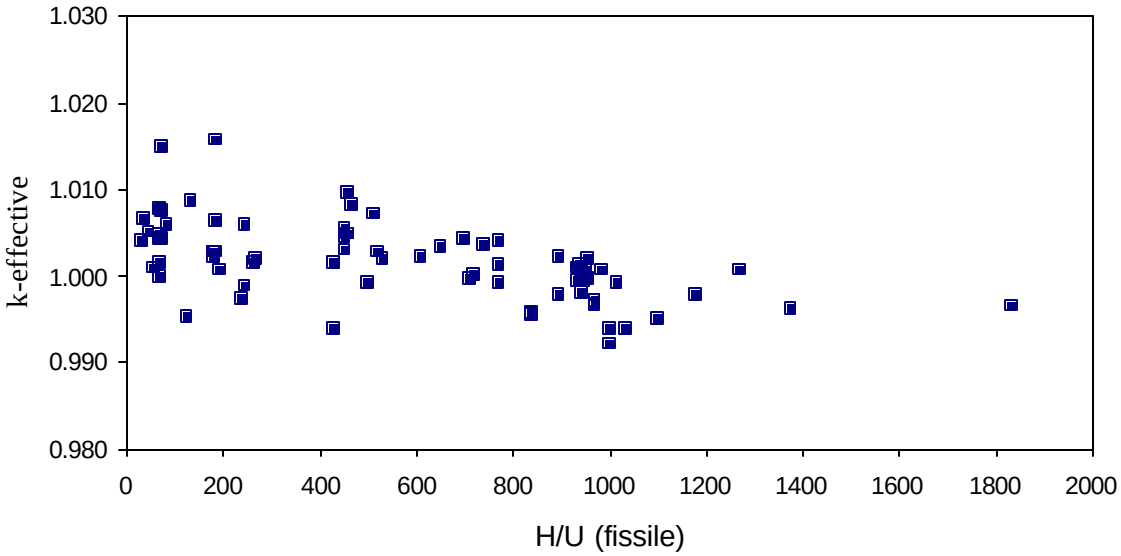
Building/System/Component	Control Mechanism	Safety Criteria
Enrichment	Enrichment	5.0 $^{w}/_{o}$ (6 $^{w}/_{o}$ ^{235}U used in NCS)
Centrifuges	Diameter	< 21.9 cm (8.6 in)
Product Cylinders (30B)	Moderation	H < 0.95 kg (2.09 lb)
Product Cylinders (48Y)	Moderation	H < 1.05 kg (2.31 lb)
UF_6 Piping	Diameter	< 21.9 cm (8.6 in)
Chemical Traps	Diameter	< 21.9 cm (8.6 in)
Product Cold Trap	Diameter	< 21.9 cm (8.6 in)
Contingency Dump System Traps	Enrichment	1.5 $^{w}/_{o}$ ^{235}U
Tanks	Mass	< 12.2 kg U (26.9 lb U)
Feed Cylinders	Enrichment	< 0.72 $^{w}/_{o}$ ^{235}U
Uranium Byproduct Cylinders	Enrichment	< 0.72 $^{w}/_{o}$ ^{235}U
UF_6 Pumps (first stage)	N/A	Safe by explicit calculation
UF_6 Pumps (second stage)	Volume	< 18.0 L (4.8 gal)
Individual Uranic Liquid Containers, e.g., Fomblin Oil Bottle, Laboratory Flask, Mop Bucket	Volume	< 18.0 L (4.8 gal)
Vacuum Cleaners Oil Containers	Volume	<18.0 L (4.8 gal)

Table 5.2-1 Uranium Solution Experiments Used for Validation

Page 1 of 1

MONK8A Case	Case Description	Number of Experiments	Handbook Reference
13	High-enriched uranyl nitrate solutions at various H:U ratios (93.17 $^{w}/_{o}$ ^{235}U)	12	HEU-SOL-THERM-002 HEU-SOL-THERM-003
23	Uranyl nitrate solution (~ 95 $^{w}/_{o}$ enriched)	5	HEU-SOL-THERM-013 NS&E
35	High-enriched uranyl nitrate solutions (U concentration from 20-700 g/L)	11	HEU-SOL-THERM-009 - HEU-SOL-THERM-012
43	Low-enriched uranyl nitrate solutions	3	LEU-SOL-THERM-002
51	Low-enriched uranium solutions (new STACY experiments)	7	LEU-SOL-THERM-004
63	Boron carbide absorber rods in uranyl nitrate (5.6 $^{w}/_{o}$ enriched)	3	LEU-SOL-THERM-005
67	Highly enriched uranyl nitrate solution with a concentration range between 59.65 and 334.66 g U/L	10	HEU-SOL-THERM-001
68	Highly enriched uranyl fluoride/heavy water solution with a concentration range between 60 and 679 g U/L and a heavy water reflector	6	HEU-SOL-THERM-004
71	STACY: 28 cm thick slabs of 10 $^{w}/_{o}$ enriched uranyl nitrate solutions, water reflected	7	LEU-SOL-THERM-016
80	STACY: Unreflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	5	LEU-SOL-THERM-007
81	STACY: Concrete reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution reflected by concrete	4	LEU-SOL-THERM-008
84	STACY: Borated concrete reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	3	LEU-SOL-THERM-009
85	STACY: Polyethylene reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	4	LEU-SOL-THERM-010

FIGURES



REFERENCE NUMBER
Figure 5.2–1.doc

FIGURE 5.2-1
VALIDATION RESULTS FOR URANIUM SOLUTIONS

REVISION DATE: DECEMBER 2003

TABLE OF CONTENTS

Page

6.0 CHEMICAL PROCESS SAFETY 6.0-1

6.1 CHEMICAL INFORMATION 6.1-1

6.1.1 Chemical Screening and Classification 6.1-1
6.1.1.1 Chemicals of Concern (Class 1) 6.1-1
6.1.1.2 Interaction Chemicals (Class 2) 6.1-2
6.1.1.3 Incidental Chemicals (Class 3) 6.1-3
6.1.2 Chemicals of Concern - Properties 6.1-3
6.1.2.1 Uranium Hexafluoride - Chemical Properties 6.1-3
6.1.2.2 Hydrogen Fluoride - Chemical Properties 6.1-4
6.1.2.3 Uranyl Fluoride - Chemical Properties 6.1-6

6.2 CHEMICAL PROCESS INFORMATION 6.2-1

6.2.1 Chemistry and Chemical Reactions 6.2-1
6.2.1.1 UF_6 and Water 6.2-1
6.2.1.2 UF_6 and Interaction Chemicals 6.2-2
6.2.1.3 UF_6 and Construction Materials 6.2-5
6.2.2 Process - General Enrichment Process 6.2-7
6.2.3 Process System Descriptions 6.2-8
6.2.4 Utility and Support System Descriptions 6.2-8
6.2.5 Safety Features 6.2-8

6.3 CHEMICAL HAZARDS ANALYSIS 6.3-1

6.3.1 Integrated Safety Analysis 6.3-1
6.3.2 Consequence Analysis Methodology 6.3-1
6.3.2.1 Defining Consequence Severity Categories 6.3-1
6.3.2.2 Chemical Release Scenarios 6.3-3
6.3.2.3 Source Term 6.3-3
6.3.2.4 Chemical Hazard Evaluation 6.3-4

6.4 CHEMICAL SAFETY ASSURANCE 6.4-1

6.4.1 Management Structure and Concepts 6.4-1
6.4.2 System Design 6.4-1
6.4.2.1 Baseline Design Criteria 6.4-2
6.4.3 Configuration Management 6.4-3
6.4.4 Maintenance 6.4-3
6.4.5 Training 6.4-4
6.4.6 Procedures 6.4-4
6.4.7 Chemical Safety Audits 6.4-6
6.4.8 Emergency Planning 6.4-6
6.4.9 Incident Investigation and Corrective Actions 6.4-7

6.5 REFERENCES 6.5-1

LIST OF TABLES

Table 6.1-1 Chemicals – Hazardous Properties
Table 6.1-2 Chemicals – Separations Building
Table 6.1-3 Chemicals – Centrifuge Assembly Building
Table 6.1-4 Chemicals – Technical Services Building
Table 6.1-5 Chemicals – Central Utilities Building
Table 6.1-6 Physical Properties of UF_6
Table 6.2-1 Properties of Chemical Adsorbents
Table 6.2-2 UF_6 Corrosion Rates
Table 6.2-3 Materials of Construction for UF_6 Systems
Table 6.3-1 ERPG and AEGL Level Definitions
Table 6.3-2 Licensed Material Chemical Consequence Categories
Table 6.3-3 ERPG and AEGL values for Hydrogen Fluoride
Table 6.3-4 ERPG and AEGL values for Uranium Hexafluoride (as soluble U)
Table 6.3-5 Enhanced Definition of Consequence Severity Categories

LIST OF FIGURES

Figure 6.1-1 UF_6 Phase Diagram
Figure 6.1-2 Densities of Solid and Liquid UF_6

6.0 CHEMICAL PROCESS SAFETY

This chapter describes the Louisiana Energy Services (LES) plan for managing chemical process safety and demonstrating that chemical process safety controls meet the requirements of 10 CFR 70 (CFR, 2003a) thereby providing reasonable assurance that the health and safety of the public and facility employees is protected. The chapter describes the chemical classification process, the hazards of chemicals of concern, process interactions with chemicals affecting licensed material and/or hazardous chemicals produced from licensed material, the methodology for evaluating hazardous chemical consequences, and the chemical safety assurance features.

The chemical process safety program for the National Enrichment Facility (NEF) is similar to attributes for chemical safety which were submitted for Nuclear Regulatory Commission (NRC) review in the LES license application for the Claiborne Enrichment Center (LES, 1993). The NRC staff evaluated these prior attributes and concluded in NUREG-1491 (NRC, 1994) that the operation of the facility would be adequately safe with respect to chemical processes and hazards.

The NEF chemical process safety program meets the acceptance criteria in Chapter 6 of NUREG-1520 (NRC, 2002) and complies with 10 CFR 70.61 (CFR, 2003b), 70.62 (CFR, 2003c) and 70.64 (CFR, 2003d).

The information provided in this chapter, the corresponding regulatory requirement and the section of NUREG-1520 (NRC, 2002) Chapter 6 in which the NRC acceptance criteria are presented are summarized below:

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 6 Reference
Section 6.1 Chemical Information		
• Properties and Hazards	70.62(c)(1)(ii)	6.4.3.1
Section 6.2 Chemical Process Information		
• General Information	70.65(b)(3)	6.4.3.1
• Design Basis, Materials, Parameters	70.62(b)	6.4.3.1
• Process Chemistry, Chemical Interaction		6.4.3.2
Section 6.3 Chemical Hazards Analysis		
• Methodology, Scenarios, Evaluation	70.65(b)(3)	6.4.3.2
Section 6.4 Chemical Safety Assurance		
• Management, Configuration Control, Design, BDC, Maintenance, Training, Procedures, Audits, Emergency Planning, Incident Investigation	70.65(b)(4)	6.4.3.2 6.4.3.3

6.1 CHEMICAL INFORMATION

This section addresses the criteria utilized to classify all site chemicals based on their potential for harm and as defined by regulatory requirements. It also presents information on the properties of those chemicals.

6.1.1 Chemical Screening and Classification

Table 6.1-1, Chemicals – Hazardous Properties, provides the listing of chemicals and related chemical wastes that are expected to be in use at the NEF. Chemical formulas in this Chapter utilize subscripting per standard convention. The hazardous properties of each chemical and related chemical waste have been listed. Also, each chemical or related waste has been classified into one of three categories (NEF Classes): Chemicals of Concern (Class 1), Interaction Chemicals (Class 2), or Incidental Chemicals (Class 3).

The definition of each classification is provided below.

Tables 6.1-2 through 6.1-5 are the basic chemical inventories for the facility. Each of these tables lists a major facility structure, area, and/or system and an associated inventory of significant chemicals/chemical usage for each area. These tables do not include the listing of all incidental sludges, wastes, and waste streams which are presented in Table 6.1-1 and do not include those chemicals that have been characterized as Class 3 materials and that are not a stored “chemical”. As such, those chemicals not included are not a process safety concern. Complete inventories of chemicals and chemical wastes (including incidental sludges, wastes, and waste streams) by area are provided in Chapter 2 of the Environmental Report.

6.1.1.1 Chemicals of Concern (Class 1)

Chemicals of Concern (NEF Class 1) are determined based on one or more characteristics of the chemical and/or the quantity in storage/use at the facility. For licensed material or hazardous chemicals produced from licensed materials, chemicals of concern are those that, in the event of release have the potential to exceed any of the concentrations defined in 10 CFR 70 (CFR, 2003a) as listed below.

<u>High Risk Chemicals of Concern</u>

1. An acute worker dose of 1 Sv (100 rem) or greater total effective dose equivalent.
2. An acute dose of 0.25 Sv (25 rem) or greater total effective dose equivalent to any individual located outside the controlled area.
3. An intake of 30 mg or greater of uranium in soluble form by any individual located outside the controlled area.

4. An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that:
   (i) Could endanger the life of a worker, or
   (ii) Could lead to irreversible or other serious, long-lasting health effects to any individual located outside the controlled area.

Intermediate Risk Chemicals of Concern

1. An acute worker dose of 0.25 Sv (25 rem) or greater total effective dose equivalent.
2. An acute dose of 0.05 Sv (5 rem) or greater total effective dose equivalent to any individual located outside the controlled area.
3. A 24-hour averaged release of radioactive material outside the restricted area in concentrations exceeding 5000 times the values in Table 2 of Appendix B to 10 CFR 20 (CFR, 2003e).
4. An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that:
   (i) Could lead to irreversible or other serious, long-lasting health effects to a worker, or
   (ii) Could cause mild transient health effects to any individual located outside the controlled area.

Non-Licensed Chemicals of Concern

For those chemicals that are not related to licensed materials, chemicals of concern are those that are listed and handled above threshold quantities of either of the following standards:

1. 29 CFR 1910.119 (CFR, 2003f) – OSHA Process Safety Management
2. 40 CFR, 68 (CFR, 2003g) – EPA Risk Management Program.

These chemicals represent, based on their inherent toxic, reactive, or flammable properties, a potential for severe chemical release and/or acute chemical exposure to an individual that:

   (i) Could endanger the life of a worker, or
   (ii) Could lead to irreversible or other serious, long-lasting health effects to any individual located outside the controlled area.

It is noted here, that uranium hexafluoride (UF_6) is the only licensed material-related chemical of concern (NEF Class 1) that will be used at the facility. There are no non-licensed chemicals of concern at the facility.

6.1.1.2 Interaction Chemicals (Class 2)

Interaction chemicals (NEF Class 2) are those chemicals/chemical systems that require evaluation for their potential to precipitate or propagate accidents in chemical of concern (NEF Class 1) systems, but by themselves are not chemicals of concern.

6.1.1.3 Incidental Chemicals (Class 3)

The facility will use other chemicals that are neither chemicals of concern nor interaction chemicals. Some of these incidental chemicals (NEF Class 3) include those that have the potential to result in injurious occupational and/or environmental exposure, but represent no potential for acute exposure to the public and which via their nature, quantity, and/or use, have no potential for impacting chemicals of concern (NEF Class 1).

These chemicals will not be subject to chemical process safety controls. Controls will be placed on incidental chemical storage, use and handling as necessary and as follows:

1. General occupational chemical safety controls will be in place for protection of facility employees in the storage, handling, and use of all chemicals as required by 29 CFR 1910 (CFR, 2003h)

2. Environmental protection controls required to prevent and/or mitigate environmental damage due to spills and discharges and to control anticipated effluents and waste are detailed in Chapter 9, Environmental Protection, and the NEF Environmental Report.

6.1.2 Chemicals of Concern - Properties

This section summarizes the chemical properties for chemicals of concern and their key byproducts.

6.1.2.1 Uranium Hexafluoride - Chemical Properties

6.1.2.1.1 Physical

Uranium hexafluoride (UF_6) is a chemical compound consisting of one atom of uranium combined with six atoms of fluorine. It is the chemical form of uranium that is used during the uranium enrichment process.

UF_6 can be a solid, liquid, or gas, depending on its temperature and pressure. Multiple phases coexist in equilibrium only under exact combinations of temperature and pressure. These properties are shown in Figure 6.1-1, UF_6 Phase Diagram, which presents the different physical forms of UF_6 as a function of temperature and pressure. The three phases are identified as regions on the diagram separated by lines representing a plot of equilibrium combinations of temperature and pressure. These boundaries all converge at one unique point on the diagram, called the triple point, where all three phases coexist in equilibrium. The triple point of UF_6 is 64°C (147°F) and 152 kPa (22 psia).

Liquid UF_6 is formed only at temperatures and pressures greater than the triple point. Below the triple point, solid UF_6 will change phase directly to UF_6 gas (sublimation) when the temperature is raised and/or the pressure is lowered at continuous points along the solid/gas interface line. This will occur without the UF_6 progressing through a liquid phase. Solid UF_6 is a white, dense, crystalline material that resembles rock salt. Both liquid and gaseous UF_6 are colorless.

Pure UF_6 follows its phase diagram consistently regardless of isotopic content. Impurities in a UF_6 cylinder will cause deviations in the normal phase behavior. The most common gaseous impurities in UF_6 feed are air and hydrogen fluoride (HF) which are generated from the reaction of UF_6 with moisture in the air. Since these light gas impurities have a higher vapor pressure than UF_6, their presence can be detected by measuring the static pressure of cylinders and comparing the results to the UF_6 phase diagram (when the UF_6 temperature is known).

UF_6 exhibits significant expansion when going from solid to liquid phase and continues to expand as the liquid temperature increases. This is illustrated in Figure 6.1-2, Densities of Solid and Liquid UF_6. This figure shows that UF_6 expands roughly 53% going from a solid at 21°C (70°F) to a liquid at 113°C (235°F). Department of Transportation cylinder fill limits are based on UF_6 density at 121°C (250°F) and provide five percent ullage or free volume as a safety factor to prevent hydraulic rupture due to heating.

Other physical properties of UF_6 are presented in Table 6.1-6, Physical Properties of UF_6.

6.1.2.1.2 Reactivity

UF_6 does not react with oxygen, nitrogen, carbon dioxide, or dry air, but it does react with water. For this reason, UF_6 is handled in leak tight containers and processing equipment. When UF_6 comes into contact with water, such as the water vapor in the air, the UF_6 and water react, forming hydrogen fluoride (HF) gas and a solid uranium-oxyfluoride compound (UO_2F_2) which is commonly referred to as uranyl fluoride. Additional information on UF_6 reactions with water is provided in Section 6.2.1, Chemistry and Chemical Reactions.

UF_6 is also incompatible with a number of other chemicals including hydrocarbons and aromatics but none of these chemicals are used in or within proximity of UF_6 process systems.

6.1.2.1.3 Toxicological

If UF_6 is released to the atmosphere, the uranium compounds and HF that are formed by reaction with moisture in the air are chemically toxic. Uranium is a heavy metal that, in addition to being radioactive, can have toxic chemical effects (primarily on the kidneys) if it enters the bloodstream by means of ingestion or inhalation. HF is an extremely corrosive gas that can damage the lungs and cause death if inhaled at high enough concentrations. Additional information on the toxicological parameters used for evaluating exposure is provided in Section 6.3, Chemical Hazards Analysis.

6.1.2.1.4 Flammability

UF_6 is not flammable and does not disassociate to flammable constituents under conditions at which it will be handled at the facility.

6.1.2.2 Hydrogen Fluoride - Chemical Properties

Hydrogen fluoride (HF) is not a direct chemical of concern (NEF Class 1), however, it is one of two byproducts of concern that would be developed in the event of most accident scenarios at

the facility. Understanding its properties therefore is important in evaluating chemical process conditions.

6.1.2.2.1 Physical

HF can exist as a gas or as a liquid under pressure (anhydrous hydrogen fluoride) or as an aqueous solution of varying strengths (aqueous hydrofluoric acid). HF vapors are colorless with a pungent odor which is detectable at concentrations above 1 ppm. It is soluble in water with a release of heat.

Releases of anhydrous hydrogen fluoride would typically fume (due to the reaction with water vapor) so that any significant release would be visible at the point of release and in the immediate vicinity.

6.1.2.2.2 Reactivity

In both gaseous and aqueous form, HF is extremely reactive, attacking certain metals, glass and other silicon-containing components, leather and natural rubber. Additional information regarding the corrosion properties and metal attack are provided in Section 6.2.1.3, UF_6 and Construction Materials.

6.1.2.2.3 Toxicological

HF in both gaseous and aqueous forms is strongly corrosive and causes severe burns to the skin, eyes and mucous membranes and severe respiratory irritation.

Inhalation of HF causes an intolerable prickling, burning sensation in the nose and throat, with cough and pain beneath the sternum. Nausea, vomiting, diarrhea and ulceration of the gums may also occur. In low concentrations, irritation of the nasal passages, dryness, bleeding from the nose and sinus disorders may result, while continued exposure can lead to ulceration and perforation of the nasal septum. Exposure to high concentrations can cause laryngitis, bronchitis and pulmonary edema which may not become apparent until 12-24 hours after the exposure.

Chronic exposure to excessive quantities of gaseous or particulate fluoride results in nausea, vomiting, loss of appetite and diarrhea or constipation. Fluorosis and other chronic effects may result from significant acute exposures. Systemic fluoride poisoning can cause hypocalcaemia which may lead to cardiac arrhythmias and/or renal failure. Chronic exposure to gaseous or particulate fluoride is not expected at the facility.

Skin exposure to concentrated liquid HF will result in aggressive chemical burns. Burns from exposure to dilute solutions (1-20%) of hydrofluoric acid (aqueous HF) or moderate concentrations of vapor may not be immediately painful or visible. Symptoms of skin exposure include immediate or delayed throbbing, burning pain followed by localized destruction of tissue and blood vessels that may penetrate to the bone. Exposure to liquid forms of HF is not expected at the facility.

Ocular exposure to HF causes a burning sensation, redness and secretion. Splashes of aqueous hydrofluoric acid to the eye rapidly produce conjunctivitis, keratitis and more serious destructive effects but these are not expected at the facility.

6.1.2.2.4 Flammability

HF is not flammable or combustible. HF can react exothermically with water to generate sufficient heat to ignite nearby combustibles. HF in reaction with certain metals can offgas hydrogen which is flammable. Both of these reactions would be more typical for bulk, concentrated HF interaction where large masses (i.e., bulk HF storage) of material are involved. These types of interactions are not expected at the facility.

6.1.2.3 Uranyl Fluoride - Chemical Properties

Uranyl fluoride (UO_2F_2) is not a direct chemical of concern (NEF Class 1), however, it is the second of two byproducts of concern (HF is the other) that would be developed in the event of a UF_6 release at the facility. Understanding its properties therefore is important in evaluating chemical process conditions.

6.1.2.3.1 Physical

UO_2F_2 is an intermediate in the conversion of UF_6 to a uranium oxide or metal form and is a direct product of the reaction of UF_6 with moisture in the air. It exists as a yellow, hygroscopic solid. UO_2F_2 formation and dispersion is governed by the conditions of the atmosphere in which the release is occurring. UF_6 will be continually hydrolyzed in the presence of water vapor. The resulting UF_6/HF cloud will include UO_2F_2 particulate matter within the gaseous stream. As this stream diffuses into larger volumes and additional UF_6 hydrolysis occurs, UO_2F_2 particulate will settle on surfaces as a solid flake-like compound. This deposition will occur within piping/equipment, on lower surfaces within enclosures/rooms, and/or on the ground – wherever the UF_6 hydrolysis reaction is occurring.

6.1.2.3.2 Reactivity

UO_2F_2 is reported to be stable in air to 300°C (570°F). It does not have a melting point because it undergoes thermal decomposition to triuranium octoxide (U_3O_8) above this temperature. When heated to decomposition, UO_2F_2 emits toxic fluoride fumes. UO_2F_2 is hygroscopic and water-soluble and will change in color from brilliant orange to yellow after reacting with water.

6.1.2.3.3 Toxicological

UO_2F_2 is radiologically and chemically toxic due to its uranium content and solubility. Once inhaled, uranyl fluoride is easily absorbed into the bloodstream because of its solubility. If large quantities are inhaled, the uranium in the uranyl complex acts as a heavy metal poison that affects the kidneys. Because of low specific activity values, the radiological toxicity of UF_6 and the UO_2F_2 byproduct are typically of less concern than the chemical toxicity.

6.1.2.3.4 Flammability

UO_2F_2 is not combustible and will not decompose to combustible constituents under conditions at which it will be handled at the facility.

6.2 CHEMICAL PROCESS INFORMATION

This section characterizes chemical reactions between chemicals of concern and interaction chemicals and other substances as applicable. This section also provides a basic discussion of some chemical processes and provides reference to Chapter 3, Integrated Safety Analysis Summary, for more detailed information on the technology, equipment, and safety systems associated with UF_6 process systems.

6.2.1 Chemistry and Chemical Reactions

Although the separation of isotopes is a physical rather than chemical process, chemical principles play an important role in the design of the facility. The phase behavior of UF_6 is critical to the design of all aspects of the plant. UF_6 has a high affinity for water and will react exothermically with water and water vapor in the air. The products of UF_6 hydrolysis, solid UO_2F_2 and gaseous HF, are both toxic. HF is also corrosive, particularly in the presence of water vapor. Because this chemical reaction results in undesirable by-products, UF_6 is isolated from moisture in the air through proper design of primary containment (i.e., piping, components, and cylinders).

Other chemical reactions occur in systems that decontaminate equipment, remove contaminants from effluent streams, and as part of lubricant recovery or other cleansing processes. Side reactions can include the corrosion and deterioration of construction materials, which influences their specification. These reactions are further described below.

6.2.1.1 UF_6 and Water

Liquid and gaseous UF_6 react rapidly with water and water vapor as does the exposed surface of solid UF_6. UF_6 reacts with water so rapidly that the HF formed is always anhydrous when in the presence of UF_6, significantly reducing its corrosive potential in cylinders, piping, and equipment. The reaction of gaseous UF_6 with water vapor at elevated temperatures is shown in Equation 6.2-1.

$$\underset{\text{(gas)}}{UF_6} + \underset{\text{(vapor)}}{2\,H_2O} \Rightarrow \underset{\text{(solid)}}{UO_2F_2} + \underset{\text{(gas)}}{4HF} + \text{heat} \qquad \text{(Eq. 6.2-1)}$$

At room temperature, depending on the relative humidity of the air, the products of this reaction are UO_2F_2 hydrates and HF- H_2O fog, which will be seen as a white cloud. A typical reaction with excess water is given in Equation 6.2-2.

$$\underset{\text{(gas)}}{UF_6} + \underset{\text{(vapor)}}{(2+4x)H_2O} \Rightarrow \underset{\text{(solid)}}{UO_2F_2 * 2\,H_2O} + \underset{\text{(fog)}}{4HF * x\,H_2O} + \text{heat} \qquad \text{(Eq. 6.2-2)}$$

If, because of extremely low humidity, the HF- H_2O fog is not formed, the finely divided uranyl fluoride (UO_2F_2) causes only a faint haze. UO_2F_2 is a water-soluble, yellow solid whose exact coloring depends on the degree of hydration as well as the particle size.

The heat release for the reaction in Equation 1 is 288.4 kJ/kg (124 BTU/lbm) of UF_6 gas reacted. The heat release is much larger if the UO_2F_2 is hydrated and HF-H_2O fog is formed with a heat release of 2,459 kJ/kg (1057 BTU/lbm) of UF_6 vapor.

These reactions, if occurring in the gaseous phase at ambient or higher temperatures, are very rapid, near instantaneous. Continuing reactions between solid UF_6 and excess water vapor occur more slowly as a uranyl fluoride layer will form on surface of the solid UF_6 which inhibits the rate of chemical reaction.

UF_6 reactions with interaction chemicals are discussed below. These include chemical reactions associated with lubricants and other chemicals directly exposed to UF_6, as well as chemicals used to recover contaminants from used lubricating oils, and capture trace UF_6, uranium compounds, and HF from effluent streams. UF_6 reactions with materials of construction are addressed in Section 6.2.1.3, UF_6 and Construction Material.

6.2.1.2 UF_6 and Interaction Chemicals

The chemistry of UF_6 is significantly affected by its fluorination and oxidation potential. Many of the chemical properties of UF_6 are attributable to the stability of the UO_2++ ion, which permits reactions with water, oxides, and salts containing oxygen-bearing anions such as SO_4--, NO_3--, and CO_3-- without liberation of the O_2 molecule.

The following subsection describes potential chemical interactions between the UF_6 process streams and interaction chemicals. Detailed descriptions of the chemical and/or utility systems utilizing interaction chemicals can be found in Chapter 3, Integrated Safety Analysis Summary.

6.2.1.2.1 PFPE (Fomblin) Oil

The reaction of UF_6 with hydrocarbons is undesirable and can be violent. Gaseous UF_6 reacts with hydrocarbons to form a black residue of uranium-carbon compounds. Hydrocarbons can be explosively oxidized if they are mixed with UF_6 in the liquid phase or at elevated temperatures. It is for this reason that non-fluorinated hydrocarbon lubricants are not utilized in any UF_6 system at the NEF.

UF_6 vacuum pumps are lubricated using PFPE (Perfluorinated Polyether) oil which is commonly referred to by a manufacturer's trade name - Fomblin oil. Fomblin oil is inert, fully fluorinated and does not react with UF_6 under any operating conditions.

Small quantities of uranium compounds and traces of hydrocarbons may be contained in the Fomblin oil used in the UF_6 vacuum pumping systems. The UF_6 degrades in the oil or reacts with trace hydrocarbons to form crystalline compounds – primarily uranyl fluoride (UO_2F_2) and uranium tetrafluoride (UF_4) particles – that gradually thicken the oil and reduce pump capacity.

Recovery of Fomblin oil for reuse in the system is conducted remotely from the UF_6 process systems. The dissolved uranium compounds are removed in a process of precipitation, centrifugation, and filtration. Anhydrous sodium carbonate (Na_2CO_3) is added to contaminated

Fomblin oil. Uranium compounds react to form sodium uranyl carbonate, which precipitates out. A filter removes the precipitate during subsequent centrifugation of the oil.

Trace amounts of hydrocarbons are then removed by adding activated carbon to the Fomblin oil and heating causing absorption of the hydrocarbons. The carbon is in turn removed through a bed of celite.

Failures associated with Fomblin oil and Fomblin oil recovery were evaluated in the Integrated Safety Analysis.

6.2.1.2.2 Chemical Traps - Activated Carbon, Aluminum Oxide, and Sodium Fluoride

Adsorption is the attraction of gas molecules to the surface of an activated solid. There are two classifications of adsorption: physical and chemical. At ordinary temperatures, adsorption is usually caused by molecular forces rather than by the formation of chemical bonds. In this type of adsorption, called physical adsorption, very little heat is evolved. If a chemical reaction takes place between the gas and the solid surface, the process is known as chemisorption. In chemisorption the reaction between surface and gas molecules occurs in a stoichiometric manner, and heat is liberated during the reaction.

Chemisorption is used in the removal of UF_6 and HF from gaseous effluent streams. It is also used to remove oil mist from vacuum pumps operating upstream of gaseous effluent ventilation systems. Adsorbent materials are placed on stationary beds in chemical traps downstream of the various cold traps. These materials capture HF and the trace amounts of UF_6 that escape desublimation during feed purification or during venting of residual UF_6 contained in hoses and/or piping that is bled down before disconnection.

The chemical traps are placed in series downstream of the cold traps in the exhaust streams to the Gaseous Effluent Vent Systems (GEVS) and may include one or more of a series of two different types of chemical traps. The first type of trap contains a charge of activated carbon to capture the small amounts of UF_6 that escape desublimation. Since chemisorption is a pressure sensitive process, HF is not fully adsorbed on carbon at low pressures. This necessitates a second type of trap containing a charge of aluminum oxide (Al_2O_3) to remove HF from the gaseous effluent stream. One or more of a series of these traps is used depending on the process system being served. Additionally, a carbon trap is present on the inlet of the vacuum pumps which discharge to the GEVS to prevent any of the pump oil from migrating back into the UF_6 cold traps.

Chemisorption of UF_6 on activated carbon evolves considerable thermal energy. This is not normally a problem in the chemical traps downstream of the cold traps because very little UF_6 escapes desublimation. If multiple equipment failures and/or operator errors occur, significant quantities of UF_6 could enter the chemical traps containing activated carbon. This could cause significant overheating leading to release. Failures associated with the carbon traps were evaluated in the Integrated Safety Analysis.

Activated carbon cannot be used in the Contingency Dump System because the relatively high UF_6 flow rates during this non-routine operation could lead to severe overheating. A chemical trap containing sodium fluoride (NaF) is installed in the contingency dump flow path to trap UF_6. NaF is used because the heat of UF_6 chemisorption on NaF is significantly lower than the heat of UF_6 chemisorption on activated carbon. Failures associated with the NaF traps were evaluated in the integrated safety analysis.

There are no specific concerns with heat of adsorption of either UF_6 or HF with Al_2O_3. Failures associated with the aluminum oxide traps were evaluated in the Integrated Safety Analysis.

The properties of these chemical adsorbents are provided in Table 6.2-1, Properties of Chemical Adsorbents.

6.2.1.2.3 Decontamination – Citric Acid

Contaminated components (e.g., pumps, valves, piping), once they are removed from the process areas, undergo decontamination. Oily parts are washed in a hot water wash that will remove the bulk of oil including residual uranic compounds. Once the hot water wash is complete, citric acid is used to remove residual uranic fluoride compound layers that are present on the component surfaces. The reaction of the uranium compounds with the citric acid solution produces various uranyl citrate complexes. After citric acid cleansing, the decontaminated component is subject to two additional water wash/rinse cycles. The entire decontamination operation is conducted in small batches on individual components.

Decontamination of sample bottles and valves is also accomplished using citric acid.

Decontamination was evaluated in the Integrated Safety Analysis. Adequate personnel protective features are in place for safely handling decontamination chemicals and byproducts.

6.2.1.2.4 Nitrogen

Gaseous nitrogen is used in the UF_6 systems for purging and filling lines that have been exposed to atmosphere for any of several reasons including: connection and disconnection of cylinders, preparing lines/components for maintenance, providing an air-excluding gaseous inventory for system vacuum pumps, and filling the interstitial space of the liquid sampling autoclave (secondary containment) prior to cylinder liquefaction.

The nitrogen system consists of a liquid nitrogen bulk storage vessel, vaporizer, gaseous nitrogen heater, liquid and gaseous nitrogen distribution lines and instrumentation. Liquid nitrogen is delivered by tanker and stored in the storage vessel.

Nitrogen is not reactive with UF_6 in any plant operational condition. Failures of the nitrogen system were evaluated in the Integrated Safety Analysis.

6.2.1.2.5 Silicone Oil

Silicone oil is used as a heat exchange medium for the heating/chilling of various cold traps. This oil is external to the UF_6 process stream in all cases and is not expected to interact with UF_6. Failures in the heating/chilling systems were evaluated in the Integrated Safety Analysis.

6.2.1.2.6 Halocarbon Refrigerants

Halocarbon refrigerants (including R23 trifluoromethane, R404A fluoromethane blend, and R507 penta/trifluoromethane) are used in individual package chillers that will provide cooling of UF_6 cylinders and/or silicon oil heat exchange media for take-off stations and cold traps. These halocarbons were selected due to good heat transfer properties, because they satisfy

environmental restrictions regarding ozone depletion, and are non-flammable. All halocarbon refrigerants are external to the UF_6 process stream in all cases and are not expected to interact with UF_6. Failures in the heating/chilling systems were evaluated in the Integrated Safety Analysis.

6.2.1.2.7 Plant Chilled Water

Chilled water is circulated in coils as a heat exchange medium for cooling of the liquid sampling autoclave after liquid samples have been drawn. Chilled water is external to the autoclave which is secondary containment for the product cylinder and sampling piping representing three physical barriers between the water and the UF_6 so no interaction is anticipated. Failures in the chilled water distribution system were evaluated in the Integrated Safety Analysis.

6.2.1.2.8 Centrifuge Cooling Water

Centrifuge cooling water is provided from the Centrifuge Cooling Water Distribution System. The function of this system is to provide a supply of deionized cooling water to the cooling coils of the centrifuges. This system provides stringent control over the operating temperature of the centrifuges to enable their efficient operation. Centrifuge cooling water is external to the UF_6 process stream in all cases and is not expected to interact with UF_6. Failures in the centrifuge cooling water distribution system were evaluated in the Integrated Safety Analysis.

6.2.1.3 UF_6 and Construction Materials

The corrosion of metallic plant components and the deterioration of non-metallic sealing materials is avoided by specifying resistant materials of construction and by controlling process fluid purity.

Direct chemical attack by the process fluid on metallic components is the result of chemical reactions. In many cases, the affinity of the process fluid for the metal produces metallic compounds, suggesting that rapid destruction of the metal would take place. This is usually prevented by the formation of a protective layer on the surface of the metal.

Deterioration of non-metallic materials is caused by exposure to process fluids and conditions. Materials used in gaskets, valves, flexible hoses, and other sealants must be sufficiently inert to have a useful service life.

UF_6 and some of its reaction products are potentially corrosive substances, particularly HF. UF_6 is a fluorinating agent that reacts with most metals. The reaction between UF_6 and metals such as nickel, copper, and aluminum produces a protective fluoride film over the metal that inhibits further reaction. These materials are therefore relatively inert to UF_6 corrosion after passivation and are suitable for UF_6 service. Aluminum is used as piping material for UF_6 systems because it is especially resistant to corrosion in the presence of UF_6. Carbon steels and stainless steels can be attacked by UF_6 at elevated temperatures but are not significantly affected by the presence of UF_6 at the operating temperatures for the facility.

Light gas impurities such as HF and air are removed from UF_6 during the purification process. Although HF is a highly corrosive substance when in solution with water as aqueous hydrofluoric acid, it contributes very little to metal corrosion when in the presence of UF_6. This is

due to the fact that UF_6 reacts with water so rapidly that HF remains anhydrous when in the presence of UF_6.

Corrosion rates of certain metals in contact with UF_6 are presented in Table 6.2-2, UF_6 Corrosion Rates, for two different temperatures. This data was provided in the original Safety Analysis Report for the Claiborne Enrichment Center (LES, 1993).

Resistant metal such as stainless steel are used in valve bellows and flex hoses. Aluminum piping is bent to minimize the use of fittings. Connections are welded to minimize the use of flanges and gaskets. As a standard practice, the use of sealant materials is minimized to reduce the number of potential leak paths.

Non-metallic materials are required to seal connections in UF_6 systems to facilitate valve and instrument replacement as well as cylinder connections. They are also used in valve packing and seating applications. All gasketing and packing material used at the facility will be confirmed as appropriate for UF_6 services. Typical materials that are resistant to UF_6 through the range of plant operating conditions include butyl rubber, Viton, and Kel-F.

The materials used to contain UF_6 are provided in Table 6.2-3, Materials of Construction for UF_6 Systems. The cylinders to be used at the facility are standard Department of Transportation approved containers for the transport and storage of UF_6, designed and fabricated in accordance with ANSI N14.1 (ANSI, applicable version). The nominal and minimum (for continued service) wall thickness for cylinders listed in Table 6.2-3, are taken from this standard.

The remaining system materials are relatively inert in the presence of UF_6 and the corrosion rates given in Table 6.2-2, indicate that these materials are acceptable for UF_6 service over the life of the plant.

As shown in Table 6.2-3, the cylinders used to store and transport UF_6 are made of carbon steel. Uranium Byproduct Cylinders (UBCs) are stored outside in open air where they are exposed to the elements. Atmospheric corrosion is determined by the exposure to moisture (e.g., rain, snow, atmospheric humidity) and the impurities in the air (such as sulfur). The corrosion rate on the outside surfaces of the carbon steel cylinders therefore varies accordingly with these conditions. Carbon steel storage cylinders are painted to provide a corrosion barrier to external elements.

External corrosion can occur on the outside cylinder surface and at interface points such as the contact point with the resting blocks and in skirt depressions (at the cylinder ends). According to a paper entitled Monitoring of Corrosion in ORGDP Cylinder Yards (DOE, 1988), the average corrosion rate experienced by UBCs is less than 0.051 mm/yr (2 mils/yr). This corrosion rate is almost exclusively due to exterior rust on the carbon steel. Another report – Prediction of External Corrosion for Steel Cylinders – 2001 Report (ORNL, 2001) – sampled exterior steel cylinders (30A) at Oak Ridge National Laboratories that had been subject to intermittent contact with the ground and found to have average corrosion rates of approximately 0.041 mm/yr (1.6 mils/yr). These values indicate that the expected service life would be greater than 50 years. These rates are conservative based on the UBC storage arrangement at the NEF. Cylinders subject to weather conditions (i.e., UBCs) will be periodically inspected to assess corrosion and corrosion rate.

6.2.2 Process - General Enrichment Process

Uranium enrichment is the process by which the isotopic composition of uranium is modified. Natural uranium consists of three isotopes, uranium 234 (^{234}U), uranium 235 (^{235}U), and uranium 238 (^{238}U), approximately 0.0058 $^{w}/_{o}$, 0.711 $^{w}/_{o}$ and 99.28 $^{w}/_{o}$ respectively. ^{235}U, unlike ^{238}U, is fissile and can sustain a nuclear chain reaction. Light water nuclear power plants (the type in the United States) normally operate on fuel containing between 2 $^{w}/_{o}$and 5 $^{w}/_{o}$ ^{235}U (low-enriched uranium); therefore, before natural uranium is used in uranium fuel for light water reactors it undergoes "enrichment."

In performing this enrichment, the NEF will receive and enrich natural uranium hexafluoride (UF_6) feed. The isotopes are separated in gas centrifuges arranged in arrays called cascades.

This process will result in the natural UF_6 being mechanically separated into two streams: (1) a product stream which is selectable up to a maximum 5 $^{w}/_{o}$ ^{235}U enrichment, and (2) a tails stream which is depleted to low percentages of ^{235}U (0.32 $^{w}/_{o}$ on average). No chemical reaction occurs during enrichment. Other processes at the plant include product blending, homogenizing and liquid sampling to ensure compliance with customer requirements and to ensure a quality product.

The enrichment process is comprised of the following major systems:

- UF_6 Feed System
- Cascade System
- Product Take-Off System
- Tails Take-Off System
- Product Blending System
- Product Liquid Sampling System.

UF_6 is delivered to the plant in ANSI N14.1 (ANSI, applicable version) standard Type 48X or 48Y international transit cylinders, which are placed in a feed station and connected to the plant via a common manifold. Heated air is circulated around the cylinder to sublime UF_6 gas from the solid phase. The gas is flow controlled through a pressure control system for distribution to the cascade system at subatmospheric pressure.

Individual centrifuges are not able to produce the desired product and tails concentration in a single step. They are therefore grouped together in series and in parallel to form arrays known as cascades. A typical cascade is comprised of many centrifuges.

UF_6 is drawn through cascades with vacuum pumps and compressed to a higher subatmospheric pressure at which it can desublime in the receiving cylinders. Highly reliable UF_6 resistant pumps will be used for transferring the process gas.

Tails material and product material are desublimed at separate chilled take-off stations. Tails material is desublimed into 48Y cylinders. Product material is desublimed into either 48Y or smaller 30B cylinders.

With the exception of liquid sampling operations, the entire enrichment process operates at subatmospheric pressure. This safety feature helps ensure that releases of UF_6 or HF are minimized because leakage would typically be inward to the system. During sampling

operations, UF_6 is liquefied within an autoclave which provides the heating required to homogenize the material for sampling. The autoclave is a rated pressure vessel which serves as secondary containment for the UF_6 product cylinders while the UF_6 is in a liquid state.

There are numerous subsystems associated with each of the major enrichment process systems as well as other facility support and utility systems. These include systems supporting venting, cooling, electrical power, air and water supply, instrumentation and control and handling functions among others.

6.2.3 Process System Descriptions

Detailed system descriptions and design information for enrichment process and process support systems are provided in Section 3.4, Enrichment And Other Process Descriptions. These descriptions include information on process technology including materials of construction, process parameters (e.g., flow, temperature, pressure, etc.), key instrumentation and control including alarms/interlocks, and items relied on for safety (IROFS).

6.2.4 Utility and Support System Descriptions

The UF_6 Enrichment Systems also interface with a number of supporting utility systems. Detailed system descriptions and design information for these utility and support systems are provided in Section 3.5, Utility and Support Systems. These descriptions include information on process technology including materials of construction; process parameters (e.g., flow, temperature, pressure, etc.), key instrumentation and control including alarms/interlocks, and (IROFS).

6.2.5 Safety Features

There are a number of safety features in place to help prevent, detect, and mitigate potential releases of UF_6. Some of these features are classified as (IROFS) as determined in the Integrated Safety Analysis (ISA). A listing of IROFS associated with process, utility and supporting systems as well as those applicable to the facility and its operations (e.g., administrative controls) is presented in Section 3.8, IROFS.

In addition to IROFS, there are other process system features that are intended to protect systems from damage that would result in an economic loss. Many of these features have a secondary benefit of enhancing safety by detecting, alarming, and/or interlocking process equipment – either prior to or subsequent to failures that result in a release of material. Some of these features are described in the individual system descriptions for each system in Chapter 3, Integrated Safety Analysis Summary.

6.3 CHEMICAL HAZARDS ANALYSIS

6.3.1 Integrated Safety Analysis

The applicant has prepared an Integrated Safety Analysis (ISA) as required under 10 CFR 70.62 (CFR, 2003c). Refer to Chapter 3, Integrated Safety Analysis Summary, for details on the ISA. As noted, the ISA:

- Provides a list of the accident sequences which have the potential to result in radiological and non-radiological releases of chemicals of concern
- Provides reasonable estimates for the likelihood and consequences of each accident identified
- Applies acceptable methods to estimate potential impacts of accidental releases.

The ISA also:

- Identifies adequate engineering and/or administrative controls (IROFS) for each accident sequence of significance
- Satisfies principles of the baseline design criteria and performance requirements in 10 CFR 70.61 (CFR, 2003b) by applying defense-in-depth to high risk chemical release scenarios
- Assures adequate levels of these controls are provided so those items relied on for safety (IROFS) will satisfactorily perform their safety functions.

The ISA demonstrates that the facility and its operations have adequate engineering and/or administrative controls in place to prevent or mitigate high and intermediate consequences from the accident sequences identified and analyzed.

6.3.2 Consequence Analysis Methodology

This section describes the methodology used to determine chemical exposure/dose and radiochemical exposure/dose criteria used to evaluate potential impact to the workers and the public in the event of material release. This section limits itself to the potential effects associated with accidental release conditions. Potential impacts from chronic (e.g., long-term) discharges from the facility are detailed in the Environmental Report.

6.3.2.1 Defining Consequence Severity Categories

The accident sequences identified by the ISA need to be categorized into one of three consequence categories (high, intermediate, or low) based on their forecast radiological, chemical, and/or environmental impacts. Section 6.1.1, Chemical Screening and Classification, presented the radiological and chemical consequence severity limits defined by 10 CFR 70.61 (CFR, 2003b) for the high and intermediate consequence categories.

To quantify criteria of 10 CFR 70.61 (CFR, 2003b) for chemical exposure, standards for each applicable hazardous chemical must be applied to determine exposure that could: (a) endanger the life of a worker; (b) lead to irreversible or other serious long-lasting health effects to an individual; and (c) cause mild transient health effects to an individual. Per NUREG-1520 (NRC 2002), acceptable exposure standards include the Emergency Response Planning Guidelines (ERPG) established by the American Industrial Hygiene Association and the Acute Exposure Guideline Levels (AEGL) established by the National Advisory Committee for Acute Guideline Levels for Hazardous Substances. The definitions of various ERPG and AEGL levels are contained in Table 6.3-1, ERPG and AEGL Level Definitions.

The consequence severity limits of 10 CFR 70.61 (CFR, 2003b) have been summarized and presented in Table 6.3-2, Licensed Material Chemical Consequence Categories. The severity limits defined in this table are developed against set criteria. Therefore, some of these limits have been further refined so that they are useful for conducting consequence analysis assessment with respect to the total dose (i.e., concentration multiplied by duration of exposure) that could reasonably be received under accident conditions.

These refinements are necessary as the chemical and radiological exposure target values are time dependent. As an example, ERPG and AEGL values for chemical exposure limits assume fixed exposure durations; these values must be appropriately scaled to exposure durations that reflect realistic exposure durations associated with a given accident.

The toxicity of UF_6 is due to its two hydrolysis products, HF and UO_2F_2. The toxicological effects of UF_6 as well as these byproducts were previously described in Section 6.1.2. AEGL and NUREG-1391 (NRC, 1991) values for HF and UF_6 were utilized for evaluation of chemotoxic exposure. Additionally, since the byproduct uranyl fluoride is a soluble uranium compound, the AEGL values were derived for evaluating soluble uranium (U) exposure in terms of both chemical toxicity and radiological dose. In general, the chemotoxicity of uranium inhalation/ingestions is of more significance than radiation dose resulting from internal U exposure. The ERPG and AEGL values for HF are presented in Table 6.3-3, ERPG and AEGL values for Hydrogen Fluoride. The ERPG and AEGL values for UF_6 (as soluble U) are presented in Table 6.3-4, ERPG and AEGL values for Uranium Hexafluoride (as soluble U).

Table 6.3-5, Enhanced Definition of Consequence Severity Categories, represents enhanced derived values as extrapolated from the HF and UF_6 (as soluble U) AEGL and NUREG-1391 (NRC, 1991) values. These enhanced definitions have been applied in order to determine consequence severity as characterized against the criteria of 10 CFR 70 (CFR, 2003a). These enhanced values have been derived using EPA recognized methodologies (NAP, 2004) for normalizing chemical exposure to values appropriate for the time intervals under consideration. The rationale associated with exposure times are further defined below.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

6.3.2.1.2 Public Exposure Assumptions

Potential exposures to members of the public were also evaluated assuming conservative assumptions for both exposure concentrations and durations. Exposure was evaluated for consequence severity against chemotoxic, radiotoxic, and radiological dose.

Public exposures were estimated to last for a duration of 30 minutes. This is consistent with self-protective criteria for UF_6/HF plumes listed in NUREG-1140 (NRC, 1988).

6.3.2.2 Chemical Release Scenarios

Section 3.7, General Types of Accident Sequences, presents all of the evaluation level chemical release scenarios based on the criteria applied in the Integrated Safety Analysis. Information on the criteria for the development of these scenarios is provided in Section 3.1, General ISA Information.

6.3.2.3 Source Term

The methodologies used to determine source term are those prescribed in NUREG/CR-6410 (NRC, 1998) and supporting documents.

6.3.2.3.1 Dispersion Methodology

In estimating the dispersion of chemical releases from the facility, conservative dispersion methodologies were utilized. Site boundary atmospheric dispersion factors were generated using a computer code based on Regulatory Guide 1.145 (NRC, 1982) methodology. The code was executed using five years (1987-1991) of meteorological data collected at Midland/Odessa, Texas, which is the closest first order National Weather Service Station to the site. This station was judged to be representative of the NEF site because the Midland Odessa National Weather Service Station site and the NEF site have similar climates and topography.

The specific modeling methods utilized follow consistent and conservative methods for source term determination, release fraction, dispersion factors, and meteorological conditions as prescribed in NRC Regulatory Guide 1.145 (NRC, 1982).

For releases inside of buildings, conservative leak path fractions were assumed as recommended by NUREG/CR-6410 (NRC, 1998) and ventilation on and off cases were evaluated for consideration of volumetric dilution and mixing efficiency prior to release to atmosphere.

6.3.2.4 Chemical Hazard Evaluation

This section is focused on presenting potential deleterious effects that might occur as a result of chemical release from the facility. As required by 10 CFR 70 (CFR, 2003a), the likelihood of these accidental releases fall into either unlikely or highly unlikely categories.

6.3.2.4.1 Potential Effects to Workers/Public

The toxicological properties of potential chemicals of concern were detailed in Section 6.2, Chemical Process Information. Section 3.7, General Types of Accident Sequences, present the evaluation level accident scenarios identified in the Integrated Safety Analysis and presents the potential consequence severities to facility workers or members of the public.

All postulated incidents have been determined to present low consequences to the workers/public, or where determined to have the potential for intermediate or high consequences, are protected with IROFS to values less than the likelihood thresholds required by 10 CFR 70.61 (CFR, 2003b).

6.3.2.4.2 Potential Effects to Facility

All postulated incidents have been determined to present inherently low consequences to the facility. No individual incident scenarios were identified that propagate additional consequence to the facility process systems or process equipment. The impact of external events on the facility, and their ability to impact process systems or equipment of concern is discussed in Section 3.7, General Types of Accident Sequences.

6.4 CHEMICAL SAFETY ASSURANCE

The facility will be designed, constructed and operated such that injurious chemical release events are prevented. Chemical process safety at the facility is assured by designing the structures, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and during any credible accident or external event.

6.4.1 Management Structure and Concepts

The criteria used for chemical process safety encompasses principles stated in NUREG-1601, Chemical Process Safety at Fuel Cycle Facilities (NRC, 1997). It is also supported by concepts advocated in 29 CFR 1910.119, Process Safety Management of Highly Hazardous Chemicals (CFR, 2003f), and 40 CFR, 68, Accidental Release Prevention Requirements (CFR, 2003g), although it is noted here that there are no chemicals at this facility which exceed threshold planning quantities of either standard.

The intent of chemical safety management principles is to identify, evaluate, and control the risk of chemical release through engineered, administrative, and related safeguards.

The chemical safety philosophy for the facility is to apply sufficient control to identify, evaluate, and control the risk of accidental chemical releases associated with licensed material production to acceptable levels in accordance with 10 CFR 70.61(b) and (c) (CFR, 2003b).

The identification and evaluation of chemical release risk has been developed through the conduct of an ISA. Credible accident scenarios as determined in the ISA have been summarized and presented in Chapter 3, Integrated Safety Analysis Summary. The development of these scenarios, and the dispersion analysis and chemical/radiological dose assessment associated with each accident sequence was performed and was conducted in accordance with NUREG/CR-6410, Nuclear Fuel Cycle Facility Accident Analysis Handbook (NRC, 1998) as was described previously in Section 6.3, Chemical Hazards Analysis.

The control of chemical release risk is ensured through numerous features that are described in the following sections.

6.4.2 System Design

The design of chemical process systems includes numerous controls for maintaining safe conditions during process operations. This is accomplished through several means including managing the arrangement and size of material containers and processes, selection and use of materials compatible with process chemicals, providing inherently safer operating conditions (e.g., vacuum handling), providing process interlocks, controls, and alarming within the chemical processes. All of these plant and equipment features help assure prevention of chemical release. Process piping and components, (e.g., centrifuges, traps, vents, etc.) are maintained safe by limits placed on their operating parameters.

6.4.2.1 Baseline Design Criteria

NUREG-1520 (NRC, 2002) requires the applicant to address baseline design criteria (BDC). With respect to chemical process safety design features recommended in NUREG-1601 (NRC, 1997), this section briefly details the features provided for the UF_6 system which is the only chemical of concern (Class 1) process system. The NEF is not proposing any facility-specific or process-specific relaxations or additions to applicable BDC features.

Details of chemical interaction between UF_6 and other chemicals (Class 2) were previously discussed in Section 6.2, Chemical Process Information. Details of the design features of all chemical process systems are provided in Chapter 3, Integrated Safety Analysis Summary. The NEF has been designed to provide adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material as required by 10 CFR 70.64 (CFR, 2003d).

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

6.4.3 Configuration Management

Configuration management includes those controls which ensure that the facility design basis is thoroughly documented and maintained, and that changes to the design basis are controlled. This includes the following:

A. That management commitment and staffing is appropriate to ensure configuration management is maintained

B. That proper quality assurance (QA) is in place for design control, document control, and records management

C. That all structures, systems, and components, including IROFS, are under appropriate configuration management.

A more detailed description of the configuration management system can be found in Section 11.1, Configuration Management (CM).

6.4.4 Maintenance

The NEF helps maintain chemical process safety through the implementation of administrative controls that ensure that process system integrity is maintained and that IROFS and other engineered controls are available and operate reliably. These controls include planned and scheduled maintenance of equipment and controls so that design features will function when required. Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance function is closely coupled to

operations. The maintenance function plans, schedules, tracks, and maintains records for maintenance activities.

Maintenance activities generally fall into the following categories:

A. Surveillance/monitoring

B. Corrective maintenance

C. Preventive maintenance

D. Functional testing.

A more detailed description of the maintenance program and maintenance management system can be found in Section 11.2, Maintenance.

6.4.5 Training

Training in chemical process safety is provided to individuals who handle licensed materials and other chemicals at the facility. The training program is developed and implemented with input from the chemical safety staff, training staff, and management. The program includes the following:

A. Analysis of jobs and tasks to determine what a worker must know to perform tasks efficiently

B. Design and development of learning objectives based upon the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker

C. Design and development of qualification requirements for positions where a level of technical capability must be achieved and demonstrated for safe and reliable performance of the job function

D. Development and implementation of standard and temporary operating procedures

E. Development and implementation of proper inspection, test, and maintenance programs and procedures

F. Development of chemical safety awareness throughout the facility so that all individuals know what their roles and responsibilities are in coordinating chemical release mitigation activities - in support of the Emergency Plan - in the event of a severe chemical release

G. Coordination of chemical process safety training curriculum with that of other areas including, radiological safety, criticality safety, facility operations, emergency response, and related areas.

A more detailed description of the training program can be found in Section 11.3, Training and Qualifications.

6.4.6 Procedures

A key element of chemical process safety is the development and implementation of procedures that help ensure reliable and safe operation of chemical process systems.

Generally, four types of plant procedures are used to control activities: operating procedures, administrative procedures, maintenance procedures, and emergency procedures.

Operating procedures, developed for workstation and Control Room operators, are used to directly control process operations. Operating procedures include:

- Directions for normal operations, including startup and some testing, operation, and shutdown, as well as off-normal conditions of operation, including alarm response
- Required actions to ensure radiological and nuclear criticality safety, chemical safety, fire protection, emergency planning, and environmental protection
- Operating limits, controls and specific direction regarding administrative controls to ensure operational safety
- Safety checkpoints such as hold points for radiological or criticality safety checks, QA verifications, or operator independent verification.

Administrative procedures are used to perform activities that support the process operations, including, but not limited to, management measures such as the following:

- Configuration management
- Nuclear criticality, radiation, chemical, and fire safety
- Quality assurance
- Design control
- Plant personnel training and qualification
- Audits and assessments
- Incident investigations
- Record keeping and document control
- Reporting.

Administrative procedures are also used for:

- Implementing the Fundamental Nuclear Material Control (FNMC) Plan
- Implementing the Emergency Plan
- Implementing the Physical Security Plan
- Implementing the Standard Practice Procedures Plan for the Protection of Classified Matter.

Maintenance procedures address:

- Preventive and corrective maintenance of IROFS
- Surveillance (includes calibration, inspection, and other surveillance testing)
- Functional testing of IROFS
- Requirements for pre-maintenance activity involving reviews of the work to be performed and reviews of procedures.

Emergency procedures address the preplanned actions of operators and other plant personnel in the event of an emergency.

A more detailed description of the procedural development and management program can be found in Section 11.4, Procedures Development and Implementation.

6.4.7 Chemical Safety Audits

Audits are conducted to determine that plant operations are performed in compliance with regulatory requirements, license conditions, and written procedures. As a minimum, they assess activities related to radiation protection, criticality safety control, hazardous chemical safety, fire protection, and environmental protection.

Audits are performed in accordance with a written plan, which identifies and schedules audits to be performed. Audit team members shall not have direct responsibility for the function and area being audited. Team members have technical expertise or experience in the area being audited and are indoctrinated in audit techniques. Audits are conducted on an annual basis on select functions and areas as defined above. The chemical process safety functions and areas will be audited at least triennially.

Qualified staff personnel that are not directly responsible for production activities are utilized to perform routine surveillances/assessments. Deficiencies noted during the inspection requiring corrective action are forwarded to the manager of the applicable area or function for action. Future surveillances/assessments include a review to evaluate if corrective actions have been effective.

A more detailed description of the audit program can be found in Section 11.5, Audits and Assessments.

6.4.8 Emergency Planning

Text removed under 10 CFR 2.390.

6.4.9 Incident Investigation and Corrective Actions

A facility wide incident investigation process exists that includes chemical process related incidents. This process is available for use by any person at the facility for reporting abnormal events and potentially unsafe conditions or activities. Abnormal events that potentially threaten or lessen the effectiveness of health, safety or environmental protection will be identified and reported to and investigated by the Health, Safety, and Environment (HS&E) Manager. 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/chemical released and/or the degree of potential for exposure of workers, the public or the environment.

A more detailed description of the incident investigation program can be found in Section 11.6, Incident Investigations and Corrective Action Process.

6.5 REFERENCES

ANSI, applicable version. Uranium Hexafluoride – Packing for Transport, American National Standards Institute, ANSI N14.1, version in effect at time of cylinder manufacture.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.62, Safety program and integrated safety analysis, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Part 20 Appendix B, Annual Limit on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage, 2003.

CFR, 2003f. Title 29, Code of Federal Regulations, Section 1910.119, Process safety management of highly hazardous chemicals, 2003.

CFR, 2003g. Title 40, Code of Federal Regulations, Part 68, Accidental Release Prevention Requirements, 2003.

CFR, 2003h. Title 29, Code of Federal Regulations, Part 1910, Occupational Safety and Health Standards, 2003.

CFR, 2004. Title 29, Code of Federal Regulations, Section 1910.120, Hazardous waste operations and emergency response, 2004.

DOE, 1988. Monitoring of Corrosion in ORGDP Cylinder Yards, presented at DOE Conference – Uranium Hexafluoride - Safe Handling, Processing, and Transporting, Henson, H.M., et al, 1988.

LES, 1993. Claiborne Enrichment Center Safety Analysis Report, Chapter 6, Louisiana Energy Services, December 1993.

NAP, 2004. “Acute Exposure Guideline Levels for Selected Airborne Chemicals,” Volume 4, National Research Council of the National Academies, National Academies Press, 2004.

NRC, 1982. Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Regulatory Guide 1.145, Revision 1, U.S. Nuclear Regulatory Commission, November 1982. (Reissued February 1983 to correct page 1.145-7)

NRC, 1988. A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees, NUREG-1140, U.S. Nuclear Regulatory Commission, January 1988.

NRC, 1991. Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation, NUREG-1391, U.S. Nuclear Regulatory Commission, February 1991.

NRC, 1994. Safety Evaluation Report for the Claiborne Enrichment Center, Homer, Louisiana, NUREG-1491, U.S. Nuclear Regulatory Commission, January 1994.

NRC, 1997. Chemical Process Safety at Fuel Cycle Facilities, NUREG-1601, U.S. Nuclear Regulatory Commission, August 1997.

NRC, 1998. Nuclear Fuel Cycle Facility Accident Analysis Handbook, NUREG/CR-6410, U.S. Nuclear Regulatory Commission, March 1998.

NRC, 2002. Standard Review Plan for the Review of a License Application for A Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

ORNL, 2001. Prediction of External Corrosion for Steel Cylinders – 2001 Report, ORNL/TM-2001/164, Oak Ridge National Laboratory, Schmoyer and Lyon, September, 2001.

TABLES

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Table 6.1-6 Physical Properties of UF_6

Page 1 of 1

Property	Value
Sublimation Point at 1.01 bar abs (14.7 psia)	56.6°C (133.8°F)
Triple Point	1.52 bar abs (22 psia) 64.1°C (147.3°F)
Density Solid @ 20°C (68°F) Liquid @ 64.1°C (147.3°F) Liquid @ 93°C (200°F) Liquid @ 113°C (235°F) Liquid @ 121°C (250°F)	 5.1 g/cc (317.8 lb/ft^3) 3.6 g/cc (227.7 lb/ft^3) 3.5 g/cc (215.6 lb/ft^3) 3.3 g/cc (207.1 lb/ft^3) 3.3 g/cc (203.3 lb/ft^3)
Heat of Sublimation @ 64.1°C (147.3°F)	135,373 J/kg (58.2 BTU/lb)
Heat of Fusion @ 64.1°C (147.3°F)	54,661 J/kg (23.5 BTU/lb)
Heat of Vaporization @ 64.1°C (147.3°F)	81,643 J/kg (35.1 BTU/lb)
Specific Heat Solid @ 27°C (81°F) Liquid @ 72°C (162°F)	 477 J/kg/°K (0.114 BTU/lb/°F) 544 J/kg/°K (0.130 BTU/lb/°F)
Critical Pressure	46.10 bar abs (668.8 psia)
Critical Temperature	230.2°C (446.4°F)

Table 6.2-1 Properties of Chemical Adsorbents

Page 1 of 1

Adsorbent (solid)/ Adsorbate (gas)	Heat of Adsorption	Capacity of Adsorption by weight
Activated Carbon/UF_6	293 kJ/kg (126 BTU/lb)	1:1
Activated Carbon/HF	negligible	negligible at low pressure
Aluminum Oxide/UF_6	negligible	0.2:1
Aluminum Oxide/HF	negligible	0.2:1
Activated NaF/UF_6	186 kJ/kg (80 BTU/lb)	1.0-1.5:1
Activated NaF/HF	4,052 kJ/kg (1,742 BTU/lb)	1:0.5

Table 6.2-2 UF_6 Corrosion Rates

Page 1 of 1

Material	Corrosion Rate @ 20°C (68°F) per year	Corrosion Rate @ 100°C (212°F) per year
Aluminum	6.6E-7 mm (2.6E-5 mils)	8.4E-5 mm (3.3E-3 mils)
Stainless Steel	1.4E-4 mm (5.5E-3 mils)	0.03 mm (1.2 mils)
Copper	1.2E-4 mm (4.7E-3 mils)	3.3E-3 mm (1.3E-1 mils)
Nickel	< 0.05 mm (< 2.0 mils)	< 0.05 mm (< 2.0 mils)

Table 6.2-3 Materials of Construction for UF_6 Systems

Page 1 of 1

Component	Material	Wall Thickness (nominal)	Wall Thickness (minimum)
UF_6 Feed Cylinders (48Y, 48X) and UBCs (48Y)	Carbon Steel ASTM A516	16 mm (0.625 inch)	12.7 mm (0.5 inch)
UF_6 Product Cylinder (30B)	Carbon Steel ASTM A516	12.7 mm (0.5 inch)	8 mm (0.3125 inch)
Sample Bottle (1S)	Nickel/Monel ASTM B162	1.6 mm (0.0625 inch)	1.6 mm (0.0625 inch)
Sample Bottle (2S)	Nickel/Monel ASTM B162	2.8 mm (0.112 inch)	1.6 mm (0.0625 inch)
UF_6 Piping	Aluminum & Stainless Steel	3.7 mm (0.147 inch)	not applicable
UF_6 Valves	Aluminum & Stainless Steel	> 3.7 mm (> 0.147 inch)	not applicable
Cold Trap	Stainless Steel	8 mm (0.315 inch)	not applicable

Table 6.3-1 ERPG and AEGL Level Definitions

Page 1 of 1

Emergency Response Planning Guideline (ERPG)		Acute Exposure Guideline Level (AEGL)	
General Definition	Values intended to provide estimates of concentration ranges above which one could be responsibly anticipate observing health effects.	**General Definition**	Threshold exposure limits for the protection of the general public, which are applicable to emergency exposure periods ranging from 10 minutes to 8 hours. It is believed that the recommended exposure levels are applicable to general population including infants and children, and other individuals who may be sensitive and susceptible.
ERPG-1	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing more than mild, transient adverse health effects or without perceiving a clearly defined objectionable odor.	**AEGL-1** (non-disabling)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience notable discomfort, irritation or certain asymptomatic, non-sensory effects. However, the effects are not disabling and are transient and reversible upon cessation of exposure.
ERPG-2	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair an individual's ability to take protective action.	**AEGL-2** (disabling)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting adverse health effects, or an impaired ability to escape.
ERPG-3	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing or developing life-threatening health effects.	**AEGL-3** (lethality)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience life-threatening health effects or death.

Table 6.3-2 Licensed Material Chemical Consequence Categories

Page 1 of 1

Consequence Categories	Workers	Offsite Public	Environment
Category 3 High	RD≥1Sv (100 rem) CD>AEGL-3, ERPG-3	RD≥0.25 Sv (25 rem) 30 mg sol U intake CD>AEGL-2, ERPG-2	
Category 2 Intermediate	RD≥0.25Sv (25 rem) CD>AEGL-2, ERPG-2	RD≥0.05 Sv (5 rem) CD>AEGL-1, ERPG-1	Radioactive release > 5000 x Table 2 Appendix B of 10 CFR Part 20
Category 1 Low	Accidents of lower radiological and chemical exposures than those above	Accidents of lower radiological and chemical exposures than those above	Radioactive releases with lower effects than those referenced above in this column

RD - Radiological Dose

CD - Chemical Dose

Table 6.3-3 ERPG and AEGL values for Hydrogen Fluoride

ERPG and AEGL Values For HF (values in mg HF/m^3)

ERPG		**AEGL**					
	1-hr		**10-min**	**30-min**	**1-hr**	**4-hr**	**8-hr**
ERPG-1	1.6	**AEGL-1**	0.8	0.8	0.8	0.8	0.8
ERPG-2	16.4	**AEGL-2**	78	28	20	9.8	9.8
ERPG-3	41	**AEGL-3**	139	51	36	18	18

Table 6.3-4 ERPG and AEGL values for Uranium Hexafluoride (as soluble U)

ERPG and AEGL Values For UF_6 (values in mg soluble U/m^3)

ERPG		AEGL					
	1-hr		**10-min**	**30-min**	**1-hr**	**4-hr**	**8-hr**
ERPG-1	3.4	**AEGL-1**	2.4	2.4	2.4	NR	NR
ERPG-2	10	**AEGL-2**	19	13	6.5	1.6	0.8
ERPG-3	20	**AEGL-3**	149	49	24	6.1	3.1

Table 6.3-5 Enhanced Definition of Consequence Severity Categories

		High Consequence (Category 3)	Intermediate Consequence (Category 2)
Acute Radiological Doses	Worker	>100 rem TEDE	>25 rem TEDE
	Outside Controlled Area	>25 rem TEDE	>5 rem TEDE
Acute Radiological Exposure	Worker	not applicable	not applicable
	Outside Controlled Area	>30 mg U intake	>5.4 mg U/m^3 (24-hr average)
Acute Chemical Exposure	Worker (local) (1-min exposure)	>40 mg U intake; > 1,300 mg HF/m^3	>10 mg U intake; >137 mg HF/m^3
	Worker (elsewhere in room) (2.5-min exposure)	Note 1 Note 2	>30 mg U/m^3 Note 2
	Worker (elsewhere in room) (5-min exposure)	>298 mg U/m^3; > 175 mg HF/m^3	>24 mg U/m^3; >98 mg HF/m^3
	Outside Controlled Area (30-min exposure)	>13 mg U/m^3; >28 mg HF/m^3	>2.4 mg U/m^3; >0.8 mg HF/m^3

Notes:

1. Use the conservative 5-minute exposure value for uranium.
2. Use the conservative 5-minute exposure value for hydrogen fluoride.

FIGURES

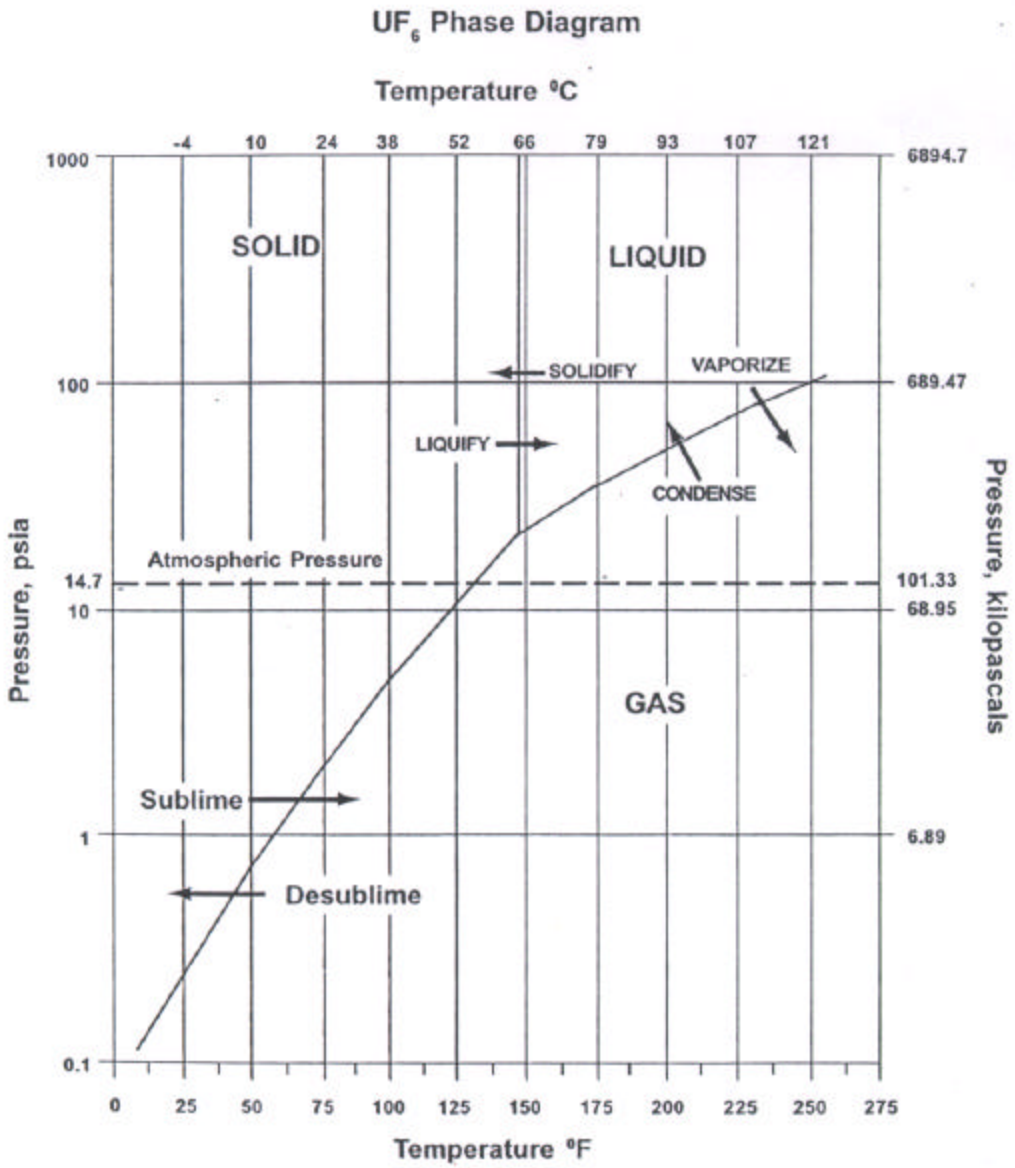


FIGURE 6.1-1

UF_6 PHASE DIAGRAM

REFERENCE NUMBER
Figures 6.1.doc

REVISION DATE: DECEMBER 2003

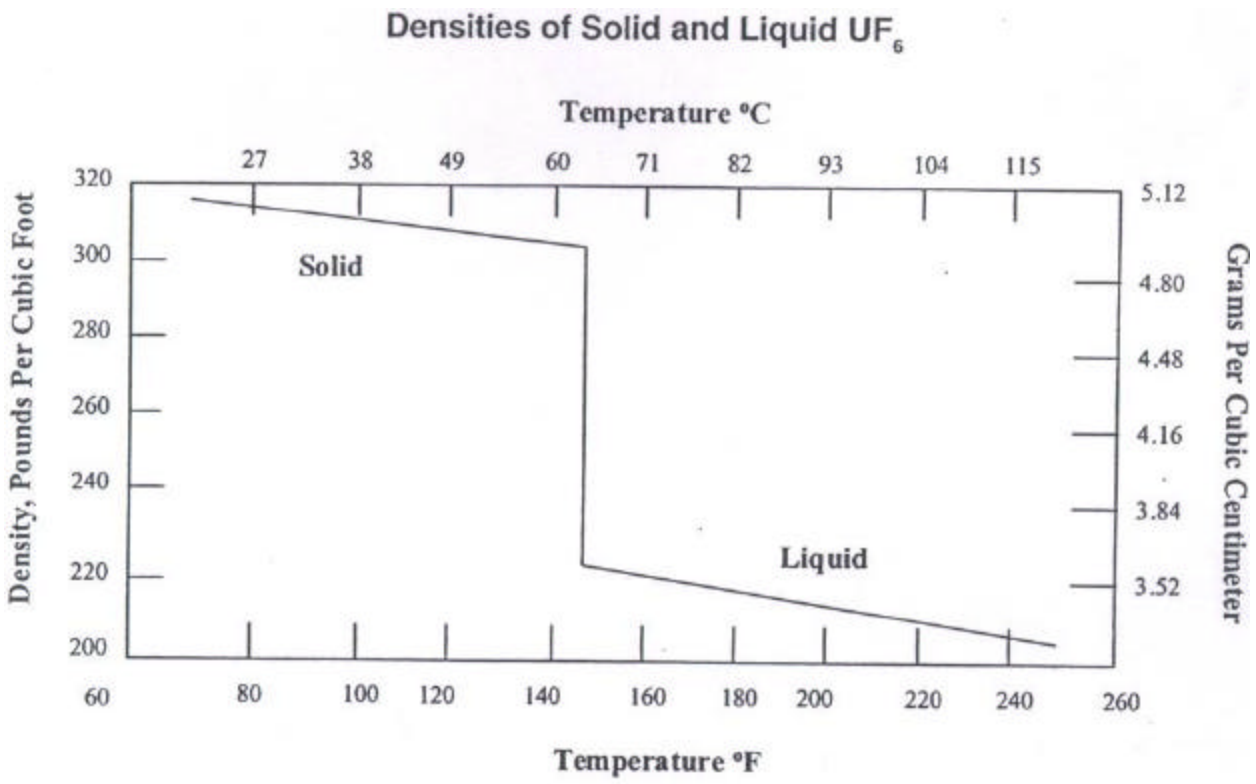


FIGURE 6.1-2

DENSITIES OF SOLID AND LIQUID UF_6

REFERENCE NUMBER

Figures 6.1.doc

REVISION DATE: DECEMBER 2003

TABLE OF CONTENTS

Page

7.0 FIRE SAFETY 7.0-1

7.1 FIRE SAFETY MANAGEMENT MEASURES 7.1-1

7.1.1 Fire Protection IROFS 7.1-1
7.1.2 Management Policy and Direction 7.1-2
7.1.3 Fire Prevention 7.1-3
7.1.4 Inspection, Testing, and Maintenance of Fire Protection Systems 7.1-3
7.1.5 Emergency Organization Qualifications, Drills and Training 7.1-4
7.1.6 Pre-Fire Plans 7.1-4

7.2 FIRE HAZARDS ANALYSIS 7.2-1

7.3 FACILITY DESIGN 7.3-1

7.3.1 Building Construction 7.3-1
7.3.2 Fire Area Determination and Fire Barriers 7.3-2
7.3.3 Electrical Installation 7.3-2
7.3.4 Life Safety 7.3-3
7.3.5 Ventilation 7.3-3
7.3.6 Drainage 7.3-4
7.3.7 Lightning Protection 7.3-4
7.3.8 Criticality Concerns 7.3-4
7.3.9 Hydrogen Control 7.3-5
7.3.10 Environmental Concerns 7.3-5
7.3.11 Physical Security Concerns 7.3-5
7.3.12 Baseline Design and Defense-In-Depth 7.3-6

7.4 PROCESS FIRE SAFETY 7.4-1

7.5 FIRE PROTECTION AND EMERGENCY RESPONSE 7.5-1

7.5.1 Fire Protection System 7.5-1
7.5.1.1 Fire Water Supply and Distribution System 7.5-1
7.5.1.2 Standpipe and Hose Systems 7.5-2
7.5.1.3 Portable Extinguishers 7.5-3
7.5.1.4 Automatic Suppression Systems 7.5-3
7.5.1.5 Fire Detection Systems 7.5-4
7.5.1.6 Manual Alarm Systems 7.5-4
7.5.1.7 Fire Alarm System 7.5-4
7.5.2 Fire Emergency Response 7.5-5
7.5.2.1 Fire Brigade 7.5-5
7.5.2.2 Off-Site Organizations 7.5-5

7.6 REFERENCES 7.6-1

LIST OF FIGURES

Figure 7.3-1 Separations Building First Floor Fire Barriers
Figure 7.3-2 Separations Building Second Floor Fire Barriers
Figure 7.3-3 Separations Building Third Floor Fire Barriers
Figure 7.3-4 Centrifuge Assembly Building First Floor Fire Barriers
Figure 7.3-5 Centrifuge Assembly Building Second Floor Fire Barriers
Figure 7.3-6 Centrifuge Assembly Building Third Floor Fire Barriers
Figure 7.3-7 Technical Services Building First Floor Fire Barriers
Figure 7.3-8 Technical Services Building Second Floor Fire Barriers
Figure 7.5-1 Exterior Fire Protection System Overall Site Plan Sheet 1 of 2
Figure 7.5-1 Exterior Fire Protection System Overall Site Plan Sheet 2 of 2
Figure 7.5-2 Sprinkler System Coverage

7.0 FIRE SAFETY

This chapter documents the National Enrichment Facility (NEF) fire safety program. The fire safety program is part of the overall facility safety program and is intended to reduce the risk of fires and explosions at the facility. The facility safety program is described in Chapter 3, Integrated Safety Analysis Summary. The fire safety program documents how the facility administers and ensures fire safety at the facility.

The NEF fire safety program meets the acceptance criteria in Chapter 7 of NUREG-1520 (NRC, 2002) and is developed, implemented and maintained in accordance with the requirements of 10 CFR 70.62(a) (CFR, 2003a), 10 CFR 70.22 (CFR, 2003b) and 10 CFR 70.65 (CFR, 2003c). In addition, the fire safety program complies with 10 CFR 70.61 (CFR, 2003d), 10 CFR 70.62 (CFR, 2003a) and 10 CFR 70.64 (CFR, 2003e). NUREG/CR-6410 (NRC, 1998), NUREG-1513 (NRC, 2001) NRC Generic Letter 95-01 (NRC, 1995) and NFPA 801 (NFPA, 2003) were utilized as guidance in developing this chapter.

The information provided in this chapter, the corresponding regulatory requirement and the section of NUREG-1520 (NRC, 2002), Chapter 7 in which the Nuclear Regulatory Commission (NRC) acceptance criteria are presented is summarized below:

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 7 Reference
Section 7.1 Fire Safety Management Measures	70.62(a), (d) & 70.64(b)	7.4.3.1
Section 7.2 Fire Hazards Analysis	70.61(b), (c) & 70.62(a)&(c)	7.4.3.2
Section 7.3 Facility Design	70.62(a), (c) & 70.64(b)	7.4.3.3
Section 7.4 Process Fire Safety	70.64(b) & 70.64(b)	7.4.3.4
Section 7.5 Fire Protection and Emergency Response	70.62(a), (c) & 70.64(b)	7.4.3.5

7.1 FIRE SAFETY MANAGEMENT MEASURES

Fire safety management measures establish the fire protection policies for the site. The objectives of the fire safety program are to prevent fires from starting and to detect, control, and extinguish those fires that do occur. The fire protection organization and fire protection systems at the NEF provide protection against fires and explosions based on the structures, systems, and components (SSC) and defense-in-depth practices described in this chapter. Fire barriers and administrative controls are considered fire protection items relied on for safety (IROFS).

7.1.1 Fire Protection IROFS

IROFS associated with fire protection are specified in Section 3.8, Items Relied on for Safety (IROFS).

7.1.2 Management Policy and Direction

Louisiana Energy Services (LES) is committed to ensuring that the IROFS, as identified in the ISA Summary, are available and reliable, and that the facility maintains fire safety awareness among employees, controls transient ignition sources and combustibles, and maintains a readiness to extinguish or limit the consequences of fire. The facility maintains fire safety awareness among employees through its General Employee Training Program. The training program is described in Chapter 11, Management Measures.

The responsibility for fire protection rests with the Health, Safety & Environment (HS&E) Manager who reports directly to the Plant Manager. The HS&E Manager is assisted by the Industrial Safety Manager, whose direct responsibility is to ensure the day-to-day safe operation of the facility in accordance with occupational safety and health regulations, including the fire safety program. Fire protection engineering support is provided by the engineering manager in Technical Services. The personnel qualification requirements for the HS&E Manager and the Industrial Safety Manager are presented in Chapter 2, Organization and Administration.

The Industrial Safety Manager is assisted by fire safety personnel who are trained in the field of fire protection and have practical day-to-day fire safety experience at nuclear facilities. The fire protection staff is responsible for the following:

- Fire protection program and procedural requirements
- Fire safety considerations
- Maintenance, surveillance, and quality of the facility fire protection features
- Control of design changes as they relate to fire protection
- Documentation and record keeping as they relate to fire protection
- Fire prevention activities (i.e., administrative controls and training)
- Organization and training of the fire brigade
- Pre-fire planning.

The facility maintains a Safety Review Committee (SRC) that reports to the Plant Manager. The SRC performs the function of a fire safety review committee. The SRC provides technical and administrative review and audit of plant operations including facility modifications to ensure that fire safety concerns are addressed.

Engineering review of the fire safety program is accomplished by configuration management and the SRC. Configuration management is discussed in Chapter 11, Management Measures, and the SRC is discussed in Chapter 2, Organization and Administration.

The subject matter discussed in Section 7.1.2 is essentially the same as the subject matter discussed in the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) relative to Management Policy and Direction (Program Management) and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on Management Policy and Direction (Program Management) is discussed in NUREG –1491 (NRC, 1994), Section 4.6.

7.1.3 Fire Prevention

Administrative controls are used to maintain the performance of the fire protection systems and delineate the responsibilities of personnel with respect to fire safety. The primary fire safety administrative controls are those that relate to fire prevention. These fire prevention controls, in the form of procedures, primarily control the storage and use of combustible materials and the use of ignition sources. These controls include, but are not limited to, the following:

- Governing the handling of transient combustibles in buildings containing IROFS, including work-generated combustibles
- Implementing a permit system to control ignition sources that may be introduced by welding, flame cutting, brazing, or soldering operations
- Ensuring that the use of open flames or combustion-generated smoke for leak testing is not permitted
- Conducting formal periodic fire prevention inspections to (1) ensure that transient combustibles adhere to established limits based on the Fire Hazard Analysis; (2) ensure the availability and acceptable condition of fire protection systems/equipment, fire stops, penetration seals, and fire-retardant coatings; and (3) ensure that prompt and effective corrective actions are taken to correct conditions adverse to fire protection and preclude their recurrence
- Performing periodic housekeeping inspections
- Implementing a permit system to control the disarming of fire detection or fire suppression systems, including appropriate compensatory measures
- Implementing fire protection system inspection, testing, and maintenance procedures.

7.1.4 Inspection, Testing, and Maintenance of Fire Protection Systems

An inspection, testing and maintenance program is implemented to ensure that fire protection systems and equipment remain operable and function properly when needed to detect and suppress fire. Fire protection procedures are written to address such topics as training of the fire brigade, reporting of fires, and control of penetration seals. The facility's Industrial Safety group has 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 penetration seals. Refer to Chapter 11, Management Measures, for additional information on procedures and maintenance activities.

The subject matter discussed in Section 7.1.4 is essentially the same as the subject matter discussed in the Claiborne Enrichment Center SAR (LES, 1993). The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) relative to Inspection, Testing, and Maintenance of Fire Protection Systems (Fire Protection Equipment Maintenance) and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on Inspection, Testing, and Maintenance of Fire Protection Systems (Fire Protection Equipment Maintenance) is discussed in NUREG –1491 (NRC, 1994), Section 4.6.

7.1.5 Emergency Organization Qualifications, Drills and Training

The qualifications, drills and training of the fire brigade members who are part of the Emergency Organization are in accordance with NFPA 600 (NFPA, 1996i). The primary purpose of the Fire Brigade Training Program is to develop a group of facility employees trained in fire prevention, fire fighting techniques, first aid procedures, and emergency response. They are trained and equipped to function as a team for the fighting of fires.

The Fire Brigade Program provides entrance and educational requirements for fire brigade candidates as well as the medical- and job-related physical requirements. The Fire Brigade Training Program provides 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.

The NEF Emergency Plan also discusses the use of offsite emergency organizations, drills and training.

7.1.6 Pre-Fire Plans

Detailed pre-fire plans will be developed for use by the facility fire brigade.

The pre-fire plans include the location of fire protection equipment, approach paths for fire response, potential hazards in the area, power supply and ventilation isolation means, important plant equipment in the area and other information considered necessary by fire emergency response personnel.

The subject matter discussed in Section 7.1.6 is essentially the same as the subject matter discussed in the Claiborne Enrichment Center SAR (LES, 1993). The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) relative to Pre-Fire Plans and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on Pre-Fire Plans is discussed in NUREG –1491 (NRC, 1994), Section 4.6.

7.2 FIRE HAZARDS ANALYSIS

A Fire Hazards Analysis (FHA) has been conducted for the facility including the fire areas and fire zones which if uncontrolled, could release UF_6 in quantity and form that could cause an intermediate or high consequence, as defined in 10 CFR 70.61 (CFR, 2003d). UF_6 is present in the Technical Services Building (TSB), Blending and Liquid Sampling Area, UF_6 Handling Area, Separations Building, Cylinder Receipt and Dispatch Building (CRDB), Centrifuge Test and Post Mortem Facilities in the Centrifuge Assembly Building (CAB) and the UBC Storage Pad.

The FHA develops bounding credible fire scenarios and then assesses the consequences of unmitigated fire.

The FHA for the facility consists of the following:

- A description of the facility's use and function
- The specific fire hazards and potential fire scenarios within the fire areas and fire zones
- The methods of consequence analysis
- The occupancy and construction requirements
- Life safety requirements
- The boundaries of the fire areas and fire zones
- The IROFS affected by the postulated fire scenarios within the fire area
- The facility response to the postulated fires
- Defense or mitigation strategy for overall facility protection.

The results of the FHA are utilized in the Integrated Safety Analysis (ISA) to identify possible fire initiators and accident sequences leading to radiological consequences or toxic chemical consequences resulting from interaction with UF_6. Chapter 3, Integrated Safety Analysis Summary, addresses the ISA.

The FHA is updated and controlled by configuration management as discussed in Chapter 11, Management Measures, to ensure that the information and analysis presented in the FHA are consistent with the current state of the facility. The FHA is reviewed and updated as necessary to incorporate significant changes and modifications to the facility, its processes, or combustible inventories.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

7.4 PROCESS FIRE SAFETY

Chapter 6, Chemical Process Safety, describes the chemical classification process, the hazards of chemicals, chemical process interactions affecting licensed material and/or hazardous chemicals produced from licensed material, the methodology for evaluating hazardous chemical consequences, and chemical safety assurance. The only process chemical of concern is uranium hexafluoride (UF_6). UF_6 is not flammable and does not disassociate to flammable constituents under conditions at which it will be handled at the NEF. The two byproducts in the event of a UF_6 release are hydrogen fluoride (HF) and uranyl fluoride (UO_2F_2) and neither presents a process fire safety hazard. Chapter 3, Integrated Safety Analysis Summary, has analyzed the hazards associated with the processes performed at the facility. The analysis did not identify any processes which represented a process fire safety hazard. Refer to Chapters 3, Integrated Safety Analysis Summary and 6, Chemical Process Safety, for additional information.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

7.6 References

ANSI / IEEE, 1974. IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations, ANSI / IEEE Std – 383-1974, American National Standards Institute / Institute of Electrical and Electronics Engineers, 1974.

ANSI, 1998. Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1998, American National Standards Institute/American Nuclear Society, 1983.

ANSI, applicable version. National Electrical Safety Code, ANSI C2, American National Standards Institute, version in effect at time of design.

ASTM, 2002. Standard Test Method for Fire Tests of Through-Penetration Fire Stops, ASTM E-814-02, American Society of Testing and Materials, 2002.

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.62, Safety program and integrated safety analysis, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 70.22, Contents of applications, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.65, Additional content of applications, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2003.

ICEA, 1986. Vertical Cable Tray Flame Tests @ 210,000 Btu, ICEA Standard T-29-520-1986, Insulated Cable Engineers Association, 1986.

IEEE, 1991. IEEE Standard for Flame Testing of Cables for Use in Cable tray in Industrial and Commercial Occupancies, IEEE 1202, Institute of Electrical and Electronics Engineers, 1991.

LES , 1993. Claiborne Enrichment Center Safety Analysis Report, Louisiana Energy Services, December 1993.

NFPA, 1993. Standard on Carbon Dioxide Extinguishing Systems, NFPA 12, National Fire Protection Association, 1993.

NFPA, 1994. Standard for Portable Fire Extinguishers, NFPA 10, National Fire Protection Association, 1994.

NFPA, 1995a. Standard for the Installation of Private Fire Service Mains and Their Appurtenances, NFPA 24, National Fire Protection Association, 1995.

NFPA, 1995b. Standard for Fire Doors and Fire Windows, NFPA 80, National Fire Protection Association, 1995.

NFPA, 1995c. Standard for Exhaust Systems for Air Conveying of Vapors, Gases, Mists, and Noncombustible Particulate Solids, NFPA 91, National Fire Protection Association, 1995.

NFPA, 1995d. Standard Methods of Tests of Fire Endurance of Building Construction and Materials, NFPA 251, National Fire Protection Association, 1995.

NFPA, 1996a. Installation of Sprinkler Systems, NFPA 13, National Fire Protection Association, 1996.

NFPA, 1996b. Standard for the Installation of Standpipe, Private Hydrants and Hose Systems, NFPA 14, National Fire Protection Association, 1996.

NFPA, 1996c. Standard for the Installation of Stationary Pumps for Fire Protection, NFPA 20, National Fire Protection Association, 1996.

NFPA, 1996d. Standard for Water Tanks for Private Fire Protection, NFPA 22, National Fire Protection Association, 1996.

NFPA, 1996e. National Electric Code, NFPA 70, National Fire Protection Association, 1996.

NFPA, 1996f. National Fire Alarm Code, NFPA 72, National Fire Protection Association, 1996.

NFPA, 1996g. Standard for the Installation of Air Conditioning and Ventilating Systems, NFPA 90A, National Fire Protection Association, 1996.

NFPA, 1996h. Standard for the Installation of Warm Air Heating and Air Conditioning Systems, NFPA 90B, National Fire Protection Association, 1996.

NFPA, 1996i. Standard on Industrial Fire Brigades, NFPA 600, National Fire Protection Association, 1996.

NFPA, 1996j. Standard on Clean Agent Fire Extinguishing Systems, NFPA 2001, National Fire Protection Association, 1996.

NFPA, 1997a. Life Safety Code, NFPA 101, National Fire Protection Association, 1997.

NFPA, 1997b. Standard for Fire Walls and Fire Barrier Walls, NFPA 221, National Fire Protection Association, 1997.

NFPA, 1997c. Standard for the Installation of Lightning Protection Systems, NFPA 780, National Fire Protection Association, 1997.

NFPA, 1999. Standard on Types of Building Construction, NFPA 220, National Fire Protection Association, 1999.

NFPA, 2000. Standard on Training for Emergency Scene Operations, NFPA 1410, National Fire Protection Association, 2000.

NFPA, 2003. Standard for Fire Protection for Facilities Handling Radioactive Materials, NFPA 801, National Fire Protection Association, 2003.

NFPA, applicable version. Standard for Electrical Safety Requirements for Employee Workplaces, NFPA 70E, National Fire Protection Association, version in effect at time of design.

NMBC, 1997. New Mexico Building Code (NMBC), Title 14, Chapter 7, Part 2, New Mexico Administrative Code, 1997.

NRC, 1994. Safety Evaluation Report for the Claiborne Enrichment Center, Homer, Louisiana, NUREG - 1491, U.S. Nuclear Regulatory Commission, January 1994.

NRC, 1995. NRC Staff Technical Position on Fire Protection for Fuel Cycle Facilities, Generic Letter 95-01, U.S. Nuclear Regulatory Commission, January 1995.

NRC, 1998. Nuclear Fuel Cycle Facility Accident Analysis Handbook, NUREG / CR-6410, U.S. Nuclear Regulatory Commission, May 1998.

NRC, 2001. Integrated Safety Analysis Guidance Document, NUREG – 1513, U.S. Nuclear Regulatory Commission, May 2001.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG - 1520, U.S. Nuclear Regulatory Commission, March 2002.

UL, 1996. Standard for High-Efficiency Particulate, Air Filter Units, UL Standard 586, 8th edition, Underwriters Laboratories, Inc., December 1996.

UL, 2001. Standard for Electrical Power and Control Tray Cables with Optional Optical-Fiber Members, UL Standard 1277, 4th edition Underwriters Laboratories, Inc., November 2001.

UL, 2003. Fire Tests of Through-Penetration Fire Stops, UL Standard 1479, 3rd edition Underwriters Laboratories, Inc., May 2003.

FIGURES

Figure removed under 10 CFR 2.390.

Figure removed under 10 CFR 2.390.

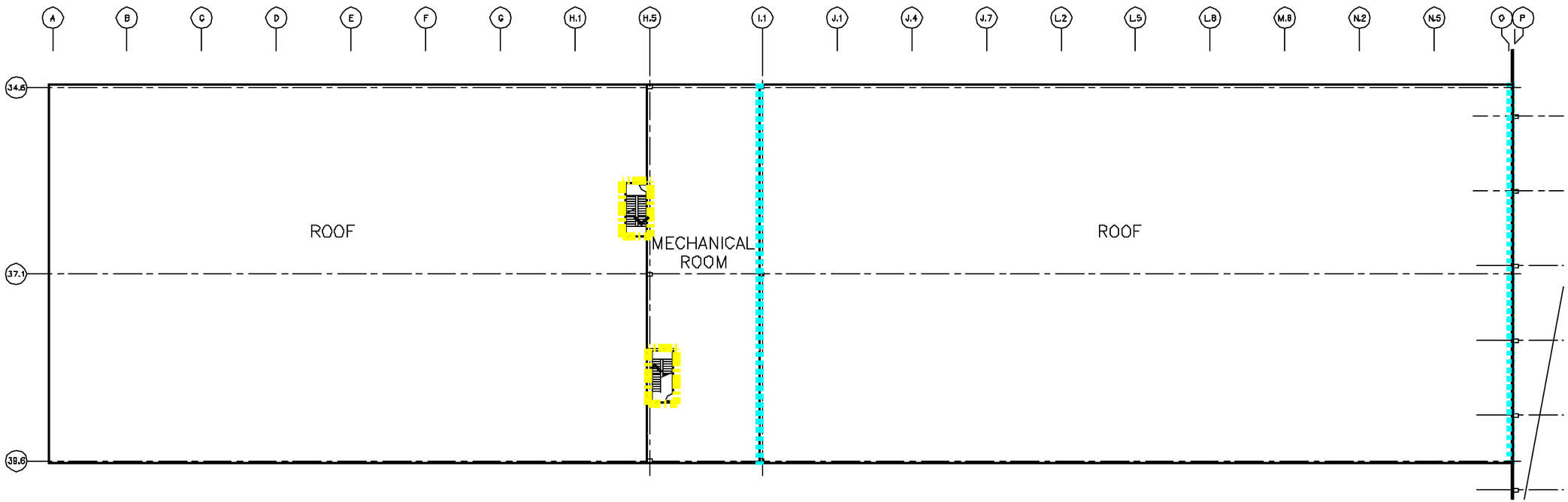
Figure removed under 10 CFR 2.390.

Figure removed under 10 CFR 2.390.

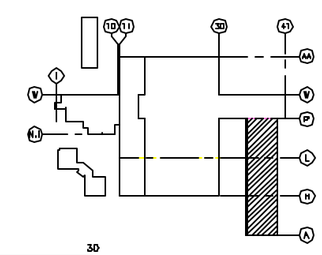
Figure removed under 10 CFR 2.390.

LEGEND

- 1-HOUR FIRE RATED PARTITION
- 2-HOUR FIRE RATED PARTITION – HORIZONTAL EXIT
- 1-HOUR FIRE RATED PARTITION (DUE TO EITHER IROFS REQUIREMENT OR OPERATIONAL CONSIDERATIONS)



ROTATED CLOCKWISE 90 DEGREES FOR CLARITY



KEYPLAN

10 0.5 0 10 20 30

SCALE 1 : 300

LOCKWOOD GREENE
ENGINEERING & CONSTRUCTION

REFERENCE NUMBER
1300A3200

FIGURE 7.3-6

CENTRIFUGE ASSEMBLY BUILDING
THIRD FLOOR FIRE BARRIERS

REVISION DATE: DECEMBER 2003

Figure removed under 10 CFR 2.390.

Figure removed under 10 CFR 2.390.

Figure removed under 10 CFR 2.390.

Figure removed under 10 CFR 2.390.

Figure removed under 10 CFR 2.390.

TABLE OF CONTENTS

Page

8.0 EMERGENCY MANAGEMENT 8.0-1

8.1 REFERENCES 8.1-1

8.0 EMERGENCY MANAGEMENT

The plans for coping with emergencies at the National Enrichment Facility are presented in the facility Emergency Plan. The Emergency Plan has been developed in accordance with 10 CFR 70.22(i) (CFR, 2003a) and 10 CFR 40.31(j) (CFR, 2003b). The Emergency Plan conforms to the guidance presented in Regulatory Guide 3.67, Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities, (NRC, 1992). The facility Emergency Plan also addresses the specific acceptance criteria in NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, (NRC, 2002), Chapter 8, Emergency Management.

The Emergency Plan identifies the offsite organizations that reviewed the Emergency Plan pursuant to the requirement in 10 CFR 70.22(i)(4) (CFR, 2003a) and 10 CFR 40.31(j)(4) (CFR, 2003b). Memorandums of Understanding with the off-site organizations are provided in the Emergency Plan.

8.1 REFERENCES

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.22, Contents of applications, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 40.31, Application for specific licenses, 2003.

NRC, 1992. Standard Format and Content of Emergency Plans for Fuel Cycle and Materials Facilities, Regulatory Guide 3.67, U.S. Nuclear Regulatory Commission, January 1992.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

TABLE OF CONTENTS

Page

9.0 ENVIRONMENTAL PROTECTION 9.0-1

9.1 ENVIRONMENTAL REPORT 9.1-1

9.1.1 Date of Application 9.1-1
9.1.2 Environmental Considerations 9.1-1
9.1.3 Analysis of Effects of Proposed Action and Alternatives 9.1-3
9.1.4 Status of Compliance 9.1-4
9.1.5 Adverse Information 9.1-4

9.2 ENVIRONMENTAL PROTECTION MEASURES 9.2-1

9.2.1 Radiation Safety 9.2-1
9.2.2 Effluent and Environmental Controls and Monitoring 9.2-2
9.2.3 Integrated Safety Analysis 9.2-2

9.3 REFERENCES 9.3-1

9.0 ENVIRONMENTAL PROTECTION

Louisiana Energy Services (LES) has prepared documents to demonstrate that its proposed environmental protective measures are adequate to protect the environment and the health and safety of the public as well as comply with the regulatory requirements imposed in 10 CFR 20 (CFR, 2003a), 10 CFR 30 (CFR, 2003b), 10 CFR 40 (CFR, 2003c), 10 CFR 51 (CFR, 2003d), and 10 CFR 70 (CFR, 2003e). The Environmental Report (ER) from LES' previous application (LES, 1994) was reviewed and information that was unchanged and found acceptable by the Nuclear Regulatory Commission (NRC) in NUREG-1484 (NRC, 1994) has been noted in the present ER.

Summarized below are the chapter section, general information category, the corresponding regulatory requirement, and the NUREG-1520 (NRC, 2002) section identifying the NRC acceptance criteria.

Chapter Section	Information Category	10 CFR Citation	NUREG-1520 Reference
9.1	**Environmental Report**	70.21(h)	9.4.3.1.1
9.1.1	Date of Application	70.21(f)	9.4.3.1.1(1)
9.1.2	Environmental Considerations	51.45(b)	9.4.3.1.1(2)
9.1.3	Analysis of Effects of Proposed Action and Alternatives	51.45(c)	9.4.3.1.1(3)
9.1.4	Status of Compliance	51.45(d)	9.4.3.1.1(4)
9.1.5	Adverse Information	51.45(e)	9.4.3.1.1(5)
9.2	**Environmental Protection Measures**	70.22(a)(8)	9.4.3.2
9.2.1	Radiation Safety	20.1101(a)	9.4.3.2.1
	• ALARA Controls and Reports	20.1101(d)	9.4.3.2.1(1)-(3)
	• Waste Minimization	20.1406	9.4.3.2.1(4)
9.2.2	Effluent and Environmental Controls and Monitoring	70.59(a)(1)	9.4.3.2.2
9.2.2.1	Effluent Monitoring	20.1501(a)	9.4.3.2.2(1)
9.2.2.2	Environmental Monitoring	20.1501(a)	9.4.3.2.2(2)
9.2.2.3	ISA Summary	70.65(b)	9.4.3.2.2(3)

This Safety Analysis Report (SAR) Chapter documents the potential environmental impacts associated with construction and operation of the NEF and indicates that adverse impacts are small. These impacts are outweighed by the substantial socioeconomic benefits associated with plant construction and operation. Additionally, the NEF will meet the underlying need for additional reliable and economical uranium enrichment capacity in the United States, thereby serving important energy and national security policy objectives. Accordingly, because the impacts of the proposed NEF are minimal and acceptable, and the benefits are desirable, the no-action alternative may be rejected in favor of the proposed action.

9.1 ENVIRONMENTAL REPORT

LES has prepared an Environmental Report (ER) that meets the requirements contained in 10 CFR Part 51 (CFR, 2003d), Subpart A. In particular, the ER addresses the requirements in 10 CFR 51.45(b)-(e) (CFR, 2003f) and follows the general format of NUREG-1748 (NRC, 2003).

The ER presents the proposed action, purpose of the proposed action, and applicable regulatory requirements (Chapter 1), discusses alternatives (Chapter 2), describes the facility and the affected environment (Chapter 3), and potential impacts of the proposed action (Chapter 4). Mitigation measures are described in Chapter 5, environmental measurements and monitoring programs in Chapter 6, a cost-benefit analysis in Chapter 7, and a summary of environmental consequences in Chapter 8. References and preparers are listed in Chapters 9 and 10, respectively.

9.1.1 Date of Application

The effective date of the ER is December 16, 2003. As required by 10 CFR 70.21(f) (CFR, 2003g), this date is at least nine months before facility construction is scheduled to begin in 2006.

9.1.2 Environmental Considerations

Applicant's ER adequately addresses the requirements of 10 CFR 51.45(b) (CFR, 2003f) as follows:

9.1.2.1 Description of the Proposed Action

The proposed action, described in ER Section 1.1, Proposed Action, is the issuance of an NRC specific license under 10 CFR 30 (CFR, 2003b), 10 CFR 40 (CFR, 2003c) and 10 CFR 70 (CFR, 2003e) to possess and use byproduct material, source material and special nuclear material (SNM) and to construct and operate a uranium enrichment facility in Lea County, New Mexico. The enriched uranium is intended for use primarily in domestic commercial nuclear power plants.

Significant characteristics of the facility are described in ER Chapters 1, Introduction of the Environmental Report and Chapter 3, Description of Affected Environment. Major site features, along with plant design and operating parameters are included. A discussion of how the special nuclear material (SNM), in this case uranium hexafluoride (UF_6), will be processed to produce enriched uranium-235 (^{235}U) is described in ER Section 1.2, Proposed Action, which also includes the proposed project schedule.

9.1.2.2 Purpose of Proposed Action

ER Section 1.2, Purpose and Need for the Proposed Action, demonstrates the need for the facility. The demonstration provides the

- Quantities of SNM used for domestic benefit

- A projection of domestic and foreign requirements for services
- Alternative sources of supply for LES' proposed services.

ER Section 1.2, Purpose and Need for the Proposed Action, also discusses if delay of the facility occurs, the effects to the nation's energy program or LES's business such as loss of contracts.

9.1.2.3 Description of the Affected Environment

Chapter 3 of the ER contains detailed descriptions of the affected environment. The chapter provides a baseline characterization of the site and its environs prior to any disturbances associated with construction or operation of the facility. The following topics and corresponding ER chapter section include:

- Site location (including longitude and latitude) and facility layout (1.2)
- Regional demography (3.10) and land use (3.1)
- Socioeconomic information (3.10), including low-income and minority populations within 130 km^2 (50 mi^2) as directed by NUREG-1748 (4.11)
- Regional historic (3.8), archeological (3.8), architectural (3.9), scenic (3.9), cultural (3.8), and natural landmarks (3.9)
- Local meteorology and air quality (3.6)
- Local surface water and ground water hydrology (3.4)
- Regional geology and seismology (3.3)
- Local terrestrial and aquatic ecology (3.5).

The baseline descriptions presented are from the most current information available. It was gathered from Federal, State, and County sources along with existing on-site data. Therefore, the information represents both seasonal and long-term environmental trends.

9.1.2.4 Discussion of Considerations

Three ER chapters discuss the potential environmental impacts relating to the proposed action. Chapter 4 details environmental and socioeconomic effects due to site preparation and facility construction and operation. Chapter 2 describes alternatives to the proposed action, including siting and designs. Chapter 7 provides a discussion of the costs and benefits for each alternative as well as the relationship between short-term use and long-term productivity of the environment, and resources committed. In addition, Chapter 8 provides a summary of environmental consequences from all actions. The associated regulatory criteria and corresponding ER section are as follows.

A. Impact of the Proposed Action on the Environment

- Effects of site preparation and construction on land (4.1) and water use (4.4)

- Effects of facility operation on human population (including consideration of occupation and public radiation exposure) and important biota (4.10, 4.11, and 4.12)

- Any irreversible commitments of resources because of site preparation and facility construction and operation, such as destruction of wildlife habitat, removal of land from agriculture, and diversion of electrical power (4.1, 7.0, and 8.2)
- Plans and policies regarding decommissioning and dismantling at the end of the facility's life (8.9)
- Environmental effects of the transportation of radioactive materials to and from the site (4.2)
- Environmental effects of accidents (4.12)
- Impacts on air (4.6) and water quality (4.4)
- Impacts on cultural and historic resources (4.8).

B. Adverse Environmental Effects

Three chapters in the ER discuss adverse environmental effects. Refer to Section 9.1.5 below for additional detail on the associated ER chapters and topics.

C. Alternatives to the Proposed Action

ER Chapter 2 provides a complete description of alternatives to the proposed action. Included are the no action alternative scenarios as well as the siting criteria and technical design requirements in sufficient detail to allow a fair and reasonable comparison between the alternatives.

D. Relationship between Short- and Long-term Productivity

ER Chapter 7, the cost-benefit analysis, included the consideration of the short-term uses and productivity of the site during the active life of the facility. No adverse impacts on the long-term productivity of the environment after decommissioning of the facility have been identified. The European experience at the Almelo enrichment plant demonstrates that a centrifuge technology site can be returned to a greenfield site for use without restriction.

E. Irreversible and Irretrievable Commitments of Resources

Irreversible environmental commitments and irretrievable material resources also are included in the cost-benefit analysis in ER Chapter 7. They are part of the capital costs associated with the land and facility and operating and maintenance costs. No significant commitments are involved with the proposed action. The site should be available for unrestricted use following decommissioning. Some components may be reused or sold as scrap during the plant life or following decommissioning.

9.1.3 Analysis of Effects of Proposed Action and Alternatives

ER Chapter 2 discusses the analysis of effects of the proposed action and alternatives in accordance with 10 CFR 51.45(c) (CFR, 2003f). The analysis considers and balances the

environmental effects of the proposed action and alternatives available to reduce or avoid both environmental and socioeconomic effects and other benefits of the proposed action.

9.1.4 Status of Compliance

ER Section 1.3 summarizes, as required in 10 CFR 51.45(d) (CFR, 2003f), the applicability of environmental regulatory requirements, permits, licenses, or approvals as well as the current status of each on the effective date of the ER.

Many federal laws and regulations apply to the facility during site assessment, construction, and operation. Some of these laws require permits from, consultations with, or approvals by, other governing or regulatory agencies. Some apply only during certain phases of facility development, rather than the entire life of the facility. Federal statutes and regulations (non-nuclear) have been reviewed to determine their applicability to the facility site assessment, construction, and operation.

9.1.5 Adverse Information

In accordance with 10 CFR 51.45(e) (CFR, 2003f), various sections throughout the ER discuss adverse environmental effects. In particular, Chapter 4 details environmental and socioeconomic effects due to site preparation and facility construction and operation. Chapter 2 compares potential impacts from alternatives. Lastly, Chapter 8 provides a summary of environmental consequences from all actions.

9.2 ENVIRONMENTAL PROTECTION MEASURES

LES is committed to protecting the public, plant workers, and the environment from the harmful effects of ionizing radiation due to plant operation. Accordingly, LES is firmly committed to the "As Low As Reasonably Achievable," (ALARA) philosophy for all operations involving source, byproduct, and special nuclear material. This commitment is reflected in written procedures and instructions for operations involving potential exposures of personnel to radiation (both internal and external hazards) and the facility design.

Part of LES's environmental protective measures are described in the ER. In particular, Chapter 4 discusses the anticipated results of the radiation protection program with regard to ALARA goals and waste minimization. Chapter 6 discusses the environmental controls and monitoring program.

A detailed description of LES' radiation protection program is included separately in this License Application as Safety Analysis Report (SAR) Chapter 4. Similarly, LES's provisions for a qualified and trained staff, which also is part of the environmental protection measures required, are described separately in the SAR as part of Chapter 11.

9.2.1 Radiation Safety

The four acceptance criteria that describe the facility radiation safety program are divided between two License Application documents. SAR Chapter 4 describes:

- Radiological (ALARA) Goals for Effluent Control
- ALARA Reviews and Reports to Management.

ER Chapter 4, Environmental Impacts, addresses:

- Effluents controls to maintain public doses ALARA, and
- Waste Minimization.

In particular, ER Section 4.12 describes public and occupational health effects from both non-radiological and radiological sources. This section specifically addresses calculated total effective dose equivalent to an average member of critical groups or calculated average annual concentration of radioactive material in gaseous and liquid effluent to maintain compliance with 10 CFR 20 (CFR, 2003a).

ER Section 4.13 contains a discussion on facility waste minimization that identifies process features and systems to reduce or eliminate waste. It also describes methods to minimize the volume of waste.

9.2.2 Effluent and Environmental Controls and Monitoring

LES has designed an environmental monitoring program to provide comprehensive data to monitor the facility's impact on the environment. The preoperational program will focus on collecting data to establish baseline information useful in evaluating changes in potential environmental conditions caused by facility operation. The preoperational program will be initiated at least two years prior to facility operation.

The operational program will monitor to ensure facility emissions are maintained ALARA. Monitoring will be of appropriate pathways up to a 2-mile radius beyond the site boundary.

ER Chapter 6 describes environmental measurement and monitoring programs as they apply to preoperation (baseline), operation, and decommissioning conditions for both the proposed action and each alternative.

9.2.2.1 Effluent Monitoring

ER Section 6.1 presents information relating to the facility radiological monitoring program. This section describes the location and characteristics of radiation sources and radioactive effluent (liquid and gaseous). It also describes the various elements of the monitoring program, including:

- Number and location of sample collection points
- Measuring devices used
- Pathway sampled or measured
- Sample size, collection frequency and duration
- Method and frequency of analysis, including lower limits of detection.

Lastly, this section justifies the choice of sample locations, analyses, frequencies, durations, sizes, and lower limits of detection.

9.2.2.2 Environmental Monitoring

ER Section 6.1 also includes information relating to the facility environmental monitoring program. The information presented is the same as that included in the effluent monitoring program, i.e., number and location of sample collection points, etc.

9.2.3 Integrated Safety Analysis

LES has prepared an integrated safety analysis (ISA) in accordance with 10 CFR 70.60 (CFR, 2003h). Refer to this SAR, Chapter 3 for a summary of the ISA. As noted, the ISA

- Provides a complete list of the accident sequences that if uncontrolled could result in radiological and non-radiological releases to the environment with intermediate or high consequences
- Provides reasonable estimates for the likelihood and consequences of each accident identified

- Applies acceptable methods to estimate environmental effects that may result from accidental releases.

The ISA also

- Identifies adequate engineering and/or administrative controls for each accident sequence of environmental significance
- Assures adequate levels are afforded so those items relied on for safety (IROFS) will satisfactorily perform their safety functions.

The ISA demonstrates that the facility and its operations have adequate engineering and/or administrative controls in place to prevent or mitigate high and intermediate consequences from the accident sequences identified and analyzed.

9.3 REFERENCES

CFR, 2003a. Title 10, Code of Federal Regulations, Part 20, Standards for Protection Against Radiation, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Part 30, Rules of General Applicability to Domestic Licensing of Byproduct Material, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Part 40, Domestic Licensing of Source Material, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Section 51.45, Environmental report, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Section 70.21, Filing, 2003.

CFR, 2003h. Title 10, Code of Federal Regulations, Section 70.60, Applicability, 2003.

LES , 1994. Claiborne Enrichment Center Environmental Report, Louisiana Energy Services, April 1994.

NRC , 1994. Final Environmental Impact Statement for the Construction and Operation of Claiborne Enrichment Center, Homer, Louisiana, NUREG-1484, Volume 1, U.S. Nuclear Regulatory Commission, August 1994.

NRC, 2003. Environmental Review Guidance for Licensing Actions Associated with NMSS Programs, Final Report , NUREG-1748, U.S. Nuclear Regulatory Commission, August 2003.

NRC, 2002. Standard Review Plan for the Review of a License Application for A Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

TABLE OF CONTENTS

Page

10.0 DECOMMISSIONING 10.0-1

10.1 SITE-SPECIFIC COST ESTIMATE 10.1-1

10.1.1 Cost Estimate Structure 10.1-1
10.1.2 Facility Description 10.1-1
10.1.3 Decommissioning Cost Estimate 10.1-1
10.1.3.1 Summary of Costs 10.1-1
10.1.3.2 Major Assumptions 10.1-2
10.1.4 Decommissioning Strategy 10.1-2
10.1.5 Decommissioning Design Features 10.1-3
10.1.5.1 Overview 10.1-3
10.1.5.2 Radioactive Contamination Control 10.1-4
10.1.5.3 Worker Exposure and Waste Volume Control 10.1-4
10.1.5.4 Management Organization 10.1-5
10.1.5.5 Health and Safety 10.1-5
10.1.5.6 Waste Management 10.1-6
10.1.5.7 Security/Material Control 10.1-6
10.1.5.8 Record Keeping 10.1-6
10.1.6 Decommissioning Process 10.1-7
10.1.6.1 Overview 10.1-7
10.1.6.2 Decontamination Facility Construction 10.1-9
10.1.6.3 System Cleaning 10.1-9
10.1.6.4 Dismantling 10.1-9
10.1.6.5 Decontamination 10.1-9
10.1.6.6 Salvage of Equipment and Materials 10.1-10
10.1.6.7 Disposal 10.1-10
10.1.6.8 Final Radiation Survey 10.1-10
10.1.7 Decontamination Facilities 10.1-11
10.1.7.1 Overview 10.1-11
10.1.7.2 Facilities Description 10.1-11
10.1.7.3 Procedures 10.1-12
10.1.7.4 Results 10.1-13
10.1.7.5 Decommissioning Impact on Integrated Safety Analysis 10.1-13

10.2 FINANCIAL ASSURANCE MECHANISM 10.2-1

10.2.1 Decommissioning Funding Mechanism 10.2-1
10.2.2 Adjusting Decommissioning Costs and Funding 10.2-1
10.2.3 Recordkeeping Plans Related to Decommissioning Funding 10.2-2

10.3 TAILS DISPOSITION 10.3-1

10.4 REFERENCES 10.4-1

LIST OF TABLES

Table 10.1-1A Number and Dimensions of Facility Components
Table 10.1-1B Number and Dimensions of Facility Components
Table 10.1-1C Number and Dimensions of Facility Components
Table 10.1-1D Number and Dimensions of Facility Components
Table 10.1-1E Number and Dimensions of Facility Components
Table 10.1-1F Number and Dimensions of Facility Components
Table 10.1-2 Planning and Preparation
Table 10.1-3 Decontamination or Dismantling of Radioactive Components
Table 10.1-4 Restoration of Contaminated Areas on Facility Grounds
Table 10.1-5 Final Radiation Survey
Table 10.1-6 Site Stabilization and Long-Term Surveillance
Table 10.1-7 Total Work Days by Labor Category
Table 10.1-8 Worker Unit Cost Schedule
Table 10.1-9 Total Labor Costs by Major Decommissioning Task
Table 10.1-10 Packaging, Shipping and Disposal of Radioactive Wastes
Table 10.1-11 Equipment and Supply Costs
Table 10.1-12 Laboratory Costs
Table 10.1-13 Period Dependent Costs
Table 10.1-14 Total Decommissioning Costs
Table 10.3-1 Summary of Depleted UF_6 Disposal Costs from Four Sources
Table 10.3-2 DOE-UDS August 29, 2002, Contract Quantities and Costs

LIST OF FIGURES

Figure 10.1-1 National Enrichment Facility – Conceptual Decommissioning Schedule

10.0 DECOMMISSIONING

This chapter presents the National Enrichment Facility (NEF) Decommissioning Funding Plan. The Decommissioning Funding Plan has been developed following the guidance provided in NUREG-1757 (NRC, 2003). This Decommissioning Funding Plan is similar to the decommissioning funding plan for the Claiborne Enrichment Center (CEC) approved by the NRC in NUREG-1491 (NRC, 1994).

Louisiana Energy Services (LES) commits to decontaminate and decommission the enrichment facility and the site at the end of its operation so that the facility and grounds can be released for unrestricted use. The Decommissioning Funding Plan will be reviewed and updated as necessary at least once every three years starting from the time of issuance of the license. Prior to facility decommissioning, a Decommissioning Plan will be prepared in accordance with 10 CFR 70.38 (CFR, 2003a) and submitted to the NRC for approval.

This chapter fulfills the applicable provisions of NUREG-1757 (NRC, 2003) through submittal of information in tabular form as suggested by the NUREG. Therefore a matrix showing compliance requirements and commitments is not provided herein.

10.1 SITE-SPECIFIC COST ESTIMATE

10.1.1 Cost Estimate Structure

The decommissioning cost estimate is comprised of three basic parts that include:

- A facility description
- The estimated costs (including labor costs, non-labor costs, and a contingency factor)
- Key assumptions.

10.1.2 Facility Description

The NEF is fully described in other sections of this License Application. Information relating to the following topics can be found in the referenced chapters listed below:

A general description of the facility and plant processes is presented in Chapter 1, General Information. A detailed description of the facility and plant processes is presented in Chapter 3, Integrated Safety Analysis Summary.

A description of the specific quantities and types of licensed materials used at the facility is provided in Chapter 1, Section 1.2, Institutional Information.

A general description of how licensed materials are used at the facility is provided in Chapter 3, Section 3.4, Enrichment and Other Process Descriptions.

10.1.3 Decommissioning Cost Estimate

10.1.3.1 Summary of Costs

The decommissioning cost estimate for the NEF is approximately $837 million (January, 2002 dollars). The decommissioning cost estimate and supporting information are presented in Tables 10.1-1A through 10.1-14, consistent with the applicable provisions of NUREG-1757, NMSS Decommissioning Standard Review Plan (NRC, 2003).

More than 97% of the decommissioning costs (except tails disposition costs) for the NEF are attributed to the dismantling, decontamination, processing, and disposal of centrifuges and other equipment in the Separations Building Modules, which are considered classified. Given the classified nature of these buildings, the data presented in the Tables at the end of this chapter has been structured to meet the applicable NUREG-1757 (NRC, 2003) recommendations, to the extent practicable. However, specific information such as numbers of components and unit rates have been intentionally excluded to protect the classified nature of the data.

The remaining 3% of the decommissioning costs are for the remaining systems and components in other buildings. Since these costs are small in relation to the overall cost estimate, the cost data for these systems has also been summarized at the same level of detail as that for the Separations Building Modules.

The decommissioning project schedule is presented in Figure 10.1-1, National Enrichment Facility – Conceptual Decommissioning Schedule. Dismantling and decontamination of the equipment in the three Separations Building Modules will be conducted sequentially (in three phases) over a nine year time frame. Separations Building Module 1 will be decommissioned during the first three-year period, followed by Separations Building Module 2, and then Separations Building Module 3. Termination of Separations Module 3 operations will mark the end of uranium enrichment operations at the NEF. Decommissioning of the remaining plant systems and buildings will begin after Separations Building Module 3 operations have been permanently terminated.

10.1.3.2 Major Assumptions

Key assumptions underlying the decommissioning cost estimate are listed below:

- Inventories of materials and wastes at the time of decommissioning will be in amounts that are consistent with routine plant operating conditions over time
- Costs are not included for the removal or disposal of non-radioactive structures and materials beyond that necessary to terminate the NRC license
- Credit is not taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning
- Decommissioning activities will be performed in accordance with current day regulatory requirements
- LES will be the Decommissioning Operations Contractor (DOC) for all decommissioning operations
- Decommissioning costs are presented in January, 2002 dollars.

10.1.4 Decommissioning Strategy

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 (i.e., immediate dismantlement), avoids long-term storage and monitoring of wastes on site. The type and volume of wastes produced at the NEF do not warrant delays in waste removal normally associated with the SAFSTOR (i.e., deferred dismantlement) option.

At the end of useful plant life, the enrichment facility will be decommissioned such that the site and remaining facilities may be released for unrestricted use as defined in 10 CFR 20.1402 (CFR, 2003b). Enrichment equipment will be removed; only building shells and the site infrastructure will remain. All remaining facilities will be decontaminated where needed to acceptable levels for unrestricted use. Confidential and Secret Restricted Data material, components, and documents will be destroyed and disposed of in accordance with the facility Standard Practice Procedures Plan for the Protection of Classified Matter.

Depleted UF_6 (tails), if not already sold or otherwise disposed of prior to decommissioning, will be disposed of in accordance with regulatory requirements. Radioactive wastes will be disposed of in licensed low-level radioactive waste disposal sites. Hazardous wastes will be treated or disposed of in licensed hazardous waste facilities. Neither tails conversion (if done), nor disposal of radioactive or hazardous material will occur at the plant site, but at licensed facilities located elsewhere.

Following decommissioning, no part of the facilities or site will remain restricted to any specific type of use.

Activities required for decommissioning have been identified, and decommissioning costs have been estimated. Activities and costs are based on actual decommissioning experience in Europe. Urenco has a fully operational dismantling and decontamination facility at its Almelo, Netherlands plant. Data and experience from this operating facility have allowed a very realistic estimation of decommissioning requirements. Using this cost data as a basis, financial arrangements are made to cover all costs required for returning the site to unrestricted use. Updates on cost and funding will be provided periodically and will include appropriate treatment for any replacement equipment. A detailed Decommissioning Plan will be submitted at a later date in accordance with 10 CFR 70.38 (CFR, 2003a).

The remaining subsections describe decommissioning plans and funding arrangements, and provide details of the decontamination aspects of the program. This information was developed in connection with the decommissioning cost estimate. Specific elements of the planning may change with the submittal of the decommissioning plan required at the time of license termination.

10.1.5 Decommissioning Design Features

10.1.5.1 Overview

Decommissioning planning begins with ensuring design features are incorporated into the plant's initial design that will simplify eventual dismantling and decontamination. The plans are implemented through proper management and health and safety programs. Decommissioning policies address radioactive waste management, physical security, and material control and accounting.

Major features incorporated into the facility design that facilitate decontamination and decommissioning are described below.

10.1.5.2 Radioactive Contamination Control

The following features primarily 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 are minimized as well.

- Certain activities during normal operation are expected to result in surface and airborne radioactive contamination. Specially designed rooms are provided for these activities to preclude contamination spread. These rooms are isolated from other areas and are provided with ventilation and filtration. The Solid Waste Collection Room, Ventilated Room and the Decontamination Workshop meet these specific design requirements.

- All areas of the plant are sectioned off into Unrestricted and Restricted Areas. Restricted Areas limit access for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Radiation Areas and Airborne Contamination Areas have additional controls to inform workers of the potential hazard in the area and to help prevent the spread of contamination. All procedures for these areas fall under the Radiation Protection Program, and serve to minimize the spread of contamination and simplify the eventual decommissioning.

- Non-radioactive process equipment and systems are minimized in locations subject to potential contamination. This limits the size of the Restricted Areas and limits the activities occurring inside these areas.

- Local air filtration is provided for areas with potential airborne contamination to preclude its spread. Fume hoods filter contaminated air in these areas.

- Curbing, pits, or other barriers are provided around tanks and components that contain liquid radioactive wastes. These serve to control the spread of contamination in case of a spill.

10.1.5.3 Worker Exposure and Waste Volume Control

The following features primarily serve to minimize worker exposure to radiation and minimize radioactive waste volumes during decontamination activities. As a result, the spread of contamination is minimized as well.

- During construction, a washable epoxy coating is applied to floors and walls that might be radioactively contaminated during operation. The coating will serve to lower waste volumes during decontamination and simplify the decontamination process. The coating is applied to floors and walls that might be radioactively contaminated during operation that are located in the Restricted Areas.

- Sealed, nonporous pipe insulation is used in areas likely to be contaminated. This will reduce waste volume during decommissioning.

- Ample access is provided for efficient equipment dismantling and removal of equipment that may be contaminated. This minimizes the time of worker exposure.

- Tanks are provided with accesses for entry and decontamination. Design provisions are also made to allow complete draining of the wastes contained in the tanks.

- Connections in the process systems provided for required operation and maintenance allow for thorough purging at plant shutdown. This will remove a significant portion of radioactive contamination prior to disassembly.

- Design drawings, produced for all areas of the plant, will simplify the planning and implementing of decontamination procedures. This in turn will shorten the durations that workers are exposed to radiation.

- Worker access to contaminated areas is controlled to assure that workers wear proper protective equipment and limit their time in the areas.

10.1.5.4 Management Organization

An appropriate organizational strategy will be developed to support the phased decommissioning schedule discussed in Section 10.1.3.1, Summary of Costs. The organizational strategy will ensure that adequate numbers of experienced and knowledgeable personnel are available to perform the technical and administrative tasks required to decommission the facility.

LES intends to be the prime Decommissioning Operations Contractor (DOC) responsible for decommissioning the NEF. In this capacity, LES will have direct control and oversight over all decommissioning activities. The role will be similar to that taken by Urenco at its facilities in Europe. In that role, Urenco has provided operational, technical, licensing, and project management support of identical facilities during both operational and decommissioning campaigns. LES also plans to secure contract services to supplement its capabilities as necessary.

Management of the decommissioning program will assure that proper training and procedures are implemented to assure worker health and safety. Programs and procedures, based on already existing operational procedures, will focus heavily on minimizing waste volumes and worker exposure to hazardous and radioactive materials. Qualified contractors assisting with decommissioning will likewise be subject to facility training requirements and procedural controls.

10.1.5.5 Health and Safety

As with normal operation, the policy during decommissioning shall be to keep individual and collective occupational radiation exposure as low as 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.

10.1.5.6 Waste Management

Radioactive and hazardous wastes produced during decommissioning will be collected, handled, and disposed of in accordance with all regulations applicable to the facility 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.

10.1.5.7 Security/Material Control

Requirements for physical security and for material control and accounting will be maintained as required during decommissioning in a manner similar to the programs in force during operation. The LES plan for completion of decommissioning, submitted near the end of plant life, will provide a description of any necessary revisions to these programs.

10.1.5.8 Record Keeping

Records important for safe and effective decommissioning of the facility will be stored in the LES Records Management System until the site is released for unrestricted use. Information maintained in these records includes:

1. Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site. These records may be limited to instances when contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as concrete. These records will include any known information on identification of involved nuclides, quantities, forms, and concentrations.

2. As-built drawings and modifications of structures and equipment in restricted areas where radioactive materials are used and/or stored and of locations of possible inaccessible contamination such as buried pipes which may be subject to contamination. Required drawings will be referenced as necessary, although each relevant document will not be indexed individually. If drawings are not available, appropriate records of available information concerning these areas and locations will be substituted.

3. Except for areas containing only sealed sources, a list contained in a single document and updated every two years, of the following:

   (i) All areas designed and formerly designated as Restricted Areas as defined under 10 CFR 20.1003; (CFR, 2003c)

   (ii) All areas outside of Restricted Areas that require documentation specified in item 1 above;

(iii) All areas outside of Restricted Areas where current and previous wastes have been buried as documented under 10 CFR 20.2108 (CFR, 2003d); and

(iv) All areas outside of Restricted Areas that contain material such that, if the license expired, the licensee would be required to either decontaminate the area to meet the criteria for decommissioning in 10 CFR 20, subpart E, (CFR, 2003e) or apply for approval for disposal under 10 CFR 20.2002 (CFR, 2003f).

4. Records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning, and records of the funding method used for assuring funds if either a funding plan or certification is used.

10.1.6 Decommissioning Process

10.1.6.1 Overview

Implementation of the DECON alternative for decommissioning may begin immediately following Separations Building Module equipment shutdown, since only low radiation levels exist at this facility. In the phased approach presented herein, dismantling and decontamination of the equipment in the three Separations Building Modules will be conducted sequentially (in three phases) over a nine year time frame. Separations Building Module 1 will be decommissioned during the first three year period, followed by Separations Building Module 2 in the next three years, and then Separations Building Module 3 in the final three years. Termination of Separations Building Module 3 operations will mark the end of uranium enrichment operations at the facility. Decommissioning of the remaining plant systems and buildings will begin after Separations Building Module 3 operations have been permanently terminated. A schematic of the NEF decommissioning schedule is presented in Figure 10.1-1, NEF – Conceptual Decommissioning Schedule.

Prior to beginning decommissioning operations, an extensive radiological survey of the facility will be performed in conjunction with a historical site assessment. The findings of the radiological survey and historical site assessment will be presented in a Decommissioning Plan to be submitted to the NRC. The Decommissioning Plan will be prepared in accordance with 10 CFR 70.38 (CFR, 2003a) and the applicable guidance provided in NUREG-1757 (NRC, 2003).

Decommissioning activities will generally include (1) installation of decontamination facilities, (2) purging of process systems, (3) dismantling and removal of equipment, (4) decontamination and destruction of Confidential and Secret Restricted Data material, (5) sales of salvaged materials, (6) disposal of wastes, and (7) completion of a final radiation survey. Credit is not taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning.

Decommissioning, using the DECON approach, requires residual radioactivity to be reduced below specified levels so the facilities may be released for unrestricted use. Current Nuclear Material Safety and Safeguards guidelines for release serve as the basis for decontamination costs estimated herein. Portions of the facility that do not exceed contamination limits may remain as is without further decontamination measures applied. The intent of decommissioning

the facility is to remove all enrichment-related equipment from the buildings such that only the building shells and site infrastructure remain. The removed equipment includes 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, cooling water and communications.

Decontamination of plant components and structures will require installation of two new facilities dedicated for that purpose. Existing plant buildings, such as the Centrifuge Assembly Building, are assumed to house the facilities. These facilities will be specially designed to accommodate repetitive cleaning of thousands of centrifuges, and to serve as a general-purpose facility used primarily for cleaning larger components. The two new facilities will be the primary location for decontamination activities during the decommissioning process. The small decontamination area in the Technical Services Building (TSB), used during normal operation, may also handle small items at decommissioning.

Decontaminated components may be reused or sold as scrap. All equipment that is to be reused or sold as scrap will be decontaminated to a level at which further use is unrestricted. Materials that cannot be decontaminated will be disposed of in a licensed radioactive waste disposal facility. As noted earlier, credit is not taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning.

Any UF_6 tails remaining on site will be removed during decommissioning. Depending on technological developments occurring prior to plant shutdown, the tails may have become marketable for further enrichment or other processes. The disposition of UF_6 tails and relevant funding provisions are discussed in Section 10.3, Tails Disposition. The cost estimate takes no credit for any value that may be realized in the future due to the potential marketability of the stored tails.

Contaminated portions of the buildings will be decontaminated as required. Structural contamination should be limited to structures in the Restricted Areas. The liners and earthen covers on the facility evaporative basins are assumed to be mildly contaminated and provisions are made for appropriate disposal of these materials in the decommissioning cost estimate. Good housekeeping practices during normal operation will maintain the other areas of the site clean.

When decontamination is complete, all areas and facilities on the site will be surveyed to verify that further decontamination is not required. Decontamination activities will continue until the entire site is demonstrated to be suitable for unrestricted use.

10.1.6.2 Decontamination Facility Construction

New facilities for decontamination can be installed in existing plant buildings to avoid unnecessary expense. Estimated time for equipment installation is approximately one year. These new facilities will be completed in time to support the dismantling and decontamination of Separations Building Module 1. These facilities are described in Section 10.1.7, Decontamination Facilities.

10.1.6.3 System Cleaning

At the end of the useful life of each Separations Building Module, the enrichment process is shut down and UF_6 is removed to the fullest extent possible by normal process operation. This is followed by evacuation and purging with nitrogen. This shutdown and purging portion of the decommissioning process is estimated to take approximately three months.

10.1.6.4 Dismantling

Dismantling is simply a matter of cutting and disconnecting all components requiring removal. The operations themselves are simple but very labor intensive. They generally require the use of protective clothing. The work process will be optimized, considering the following.

- Minimizing the spread of contamination 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
- Balancing the cost of decontamination and salvage with the cost of disposal.

Details of the complex optimization process will necessarily be decided near the end of plant life, taking into account specific contamination levels, market conditions, and available waste disposal sites. To avoid laydown space and contamination problems, dismantling should be allowed to proceed generally no faster than the downstream decontamination process. The time frame to accomplish both dismantling and decontamination is estimated to be approximately three years per Separations Building Module.

10.1.6.5 Decontamination

The decontamination process is addressed separately in detail in Section 10.1.7.

10.1.6.6 Salvage of Equipment and Materials

Items to be removed from the facilities can be categorized as potentially re-usable equipment, recoverable scrap, and wastes. However, based on a 30 year facility operating license, operating equipment is not assumed to have reuse value. Wastes will also have no salvage value.

With respect to scrap, a significant amount of aluminum will be recovered, along with smaller amounts of steel, copper, and other metals. For security and convenience, the uncontaminated materials will likely be smelted to standard ingots, and, if possible, sold at market price. The contaminated materials will be disposed of as low-level radioactive waste. No credit is taken for any salvage value that might be realized from the sale of potential assets during or after decommissioning.

10.1.6.7 Disposal

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. Wastes will consist of normal industrial trash, non-hazardous chemicals and fluids, small amounts of hazardous materials, and radioactive wastes. The radioactive waste will consist primarily of crushed centrifuge rotors, trash, and citric cake. Citric cake consists of uranium and metallic compounds precipitated from citric acid decontamination solutions. It is estimated that approximately 5,000 m^3 (6,539 yd^3) of radioactive waste will be generated over the nine-year decommissioning operations period. (This waste is subject to further volume reduction processes prior to disposal).

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 effluent to be produced during decommissioning will be provided in the Decommissioning Plan that will be submitted prior to initiating the decommissioning of the plant.

Confidential and Secret Restricted Data components and documents on site shall be disposed of in accordance with the requirements of 10 CFR 95 (CFR, 2003g). Such 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. Details will be provided in the facility Standard Practice Procedures Plan for the Protection of Classified Matter and Information, submitted separately in accordance with 10 CFR 95 (CFR, 2003g).

10.1.6.8 Final Radiation Survey

A final radiation survey must be performed to verify proper decontamination to allow the site to be released for unrestricted use. The evaluation of the final radiation survey is based in part on an initial radiation survey performed prior to initial operation. The initial survey determines the

natural background radiation of the area; therefore it provides a datum for measurements which determine any increase in levels of radioactivity.

The final survey will systematically measure radioactivity over the entire site. The intensity of the survey will vary depending on the location (i.e. the buildings, the immediate area around the buildings, 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. The results will be analyzed and shown to be below allowable residual radioactivity limits; otherwise, further decontamination will be performed.

10.1.7 Decontamination Facilities

10.1.7.1 Overview

The facilities, procedures, and expected results of decontamination are described in the paragraphs below. Since reprocessed uranium will not be used as feed in the NEF, no consideration of ^{232}U, transuranic alpha-emitters and fission product residues is necessary for the decontamination process. Only contamination from ^{238}U, ^{235}U, ^{234}U, and their daughter products will require handling by decontamination processes. The primary contaminant 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.

10.1.7.2 Facilities Description

A decontamination facility will be required to accommodate decommissioning. This 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 required for handling the remainder of the various plant components. These facilities are assumed to be installed in existing plant buildings (such as the Centrifuge Assembly Building).

The decontamination facility will have four functional areas that include (1) a disassembly area, (2) a buffer stock area, (3) a decontamination area, and (4) a scrap storage area for cleaned stock. The general purpose facility may share the specialized 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. Barriers and other physical measures will be installed and administrative controls implemented, as needed, to limit the spread of contamination.

Equipment in the decontamination facility is assumed to include:

- Transport and manipulation equipment
- Dismantling tables for centrifuge externals
- Sawing machines

- Dismantling boxes and tanks, for centrifuge internals
- Degreasers
- Citric acid and demineralized water baths
- Contamination monitors
- Wet blast cabinets
- Crusher, for centrifuge rotors
- Smelting and/or shredding equipment
- Scrubbing facility.

The decontamination facilities provided in the TSB for normal operational needs would also be available for cleaning small items during decommissioning.

10.1.7.3 Procedures

Formal procedures for all major decommissioning activities 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. The experience of decommissioning European gas centrifuge enrichment facilities will be incorporated extensively into the procedures.

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. If they are accidentally contaminated during normal operation, they would be cleaned up when the contamination is discovered. This limits the scope of necessary decontamination at the time of decommissioning.

Contaminated plant components will be cut up or dismantled, then processed through the decontamination facilities. Contamination of site structures will be limited to areas in the Separations Building Modules and TSB, and will be maintained at low levels throughout plant operation by regular cleaning. The Decontamination Workshop Area, Ventilated Room, Vacuum Pump Rebuild Workshop, and a portion of the Laundry Room are included as permanent Restricted Areas. Through the application of special protective coatings, to surfaces that might become radioactively contaminated during operation, and good housekeeping practices, final decontamination of these areas is assumed to require minimal removal of surface concrete or other structural material.

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.

10.1.7.4 Results

Urenco plant experience in Europe has demonstrated that conventional decontamination techniques are effective for all plant items. Recoverable items have been decontaminated and made suitable for reuse except for a very small amount of intractably contaminated material. The majority of radioactive waste requiring disposal in the NEF will include crushed centrifuge rotors, trash, and residue from the effluent treatment systems.

European experience has demonstrated that the aluminum centrifuge casings can be successfully decontaminated and recycled. However, as a conservative measure for this decommissioning cost estimate, the aluminum centrifuge casings for the NEF are assumed to be disposed of as low-level radioactive waste.

Overall, no problems are anticipated that will prevent the site from being released for unrestricted use.

10.1.7.5 Decommissioning Impact on Integrated Safety Analysis (ISA)

As was described in Section 10.1.3.1, Summary of Costs, dismantling and decontamination of the equipment in the three Separations Building Modules will be conducted sequentially (in three phases) over a nine year time frame. Separations Building Module 1 will be decommissioned during the first three-year period, followed by Separations Building Module 2, and then Separations Building Module 3. Termination of Separations Module 3 operations will mark the end of uranium enrichment operations at the NEF. Decommissioning of the remaining plant systems and buildings will begin after Separations Building Module 3 operations have been permanently terminated.

Although decommissioning operations are planned to be underway while all the activities considered in the ISA continue to occur in the other portions of the plant, the current ISA has not considered these decommissioning risks. An updated ISA will be performed at a later date, but prior to decommissioning, to incorporate the risks from decommissioning operations on concurrent enrichment operations.

10.2 FINANCIAL ASSURANCE MECHANISM

10.2.1 Decommissioning Funding Mechanism

LES intends to utilize a surety method, such as a letter or line of credit or surety bond, to provide reasonable assurance of decommissioning funding as required by 10 CFR 40.36(e)(2) (CFR, 2003h) and 70.25(f)(2) (CFR, 2003i). Finalization of the specific financial instruments to be utilized will be completed, and signed originals of those instruments will be provided to the NRC, prior to LES receipt of licensed material. LES intends to provide continuous financial assurance from the time of receipt of licensed material to the completion of decommissioning and termination of the license. Since LES intends to sequentially install and operate the Separations Building Modules over time, financial assurance for decommissioning will be provided during the operating life of the NEF at a rate that is in proportion to the decommissioning liability for these facilities as they are phased in. Similarly, LES will provide decommissioning funding assurance for disposition of depleted tails at a rate in proportion to the amount of accumulated tails onsite up to the maximum amount of the tails as described in Section 10.3, Tails Disposition.

The surety method adopted by LES will provide an ultimate guarantee that decommissioning costs will be paid in the event LES is unable to meet its decommissioning obligations at the time of decommissioning. The surety method will also be structured and adopted consistent with applicable NRC regulatory requirements and in accordance with NRC regulatory guidance contained in NUREG-1757 (NRC, 2003). Accordingly, LES intends that its surety method will contain, but not be limited to, the following attributes:

- The surety method will be open-ended or, if written for a specified term, such as five years, will be renewed automatically unless 90 days or more prior to the renewal date, the issuer notifies the NRC, the trust to which the surety is payable, and LES of its intention not to renew. The surety method will also provide that the full face amount be paid to the beneficiary automatically prior to the expiration without proof of forfeiture if LES fails to provide a replacement acceptable to the NRC within 30 days after receipt of notification of cancellation.

- The surety method will be payable to a trust established for decommissioning costs. The trustee and trust will be ones acceptable to the NRC. For instance, the trustee may be an appropriate State or Federal government agency or an entity which has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency.

- The surety method will remain in effect until the NRC has terminated the license.

10.2.2 Adjusting Decommissioning Costs and Funding

In accordance with 10 CFR 40.36(d) (CFR, 2003h) and 70.25(e) (CFR, 2003i), LES will update the decommissioning cost estimate for the NEF, and the associated funding levels, over the life

of the facility. These updates will take into account changes resulting from inflation or site-specific factors, such as changes in facility conditions or expected decommissioning procedures. These funding level updates will also address anticipated operation of additional Separations Building Modules and accumulated tails.

As required by the applicable regulations 10 CFR 70.25(e) (CFR, 2003i), such updating will occur approximately every three years. A record of the update process and results will be retained for review as discussed in Section 10.2.3, below. The NRC will be notified of any material changes to the decommissioning cost estimate and associated funding levels (e.g., significant increases in costs beyond anticipated inflation). To the extent the underlying instruments are revised to reflect changes in funding levels, the NRC will be notified as appropriate.

10.2.3 Recordkeeping Plans Related to Decommissioning Funding

In accordance with 10 CFR 40.36(f) (CFR, 2003h) and 70.25(g) (CFR, 2003i), LES will retain records, until the termination of the license, of information that could have a material effect on the ultimate costs of decommissioning. These records will include information regarding: (1) spills or other contamination that cause contaminants to remain following cleanup efforts; (2) as-built drawings of structures and equipment, and modifications thereto, where radioactive contamination exists (e.g., from the use or storage of such materials); (3) original and modified cost estimates of decommissioning; and (4) original and modified decommissioning funding instruments and supporting documentation.

10.3 TAILS DISPOSITION

The disposition of tails from the NEF is an element of authorized operating activities. It involves neither decommissioning waste nor is it a part of decommissioning activities. The disposal of these tails is analogous to the disposal of radioactive materials generated in the course of normal operations (even including spent fuel in the case of a power reactor), which is authorized by the operating license and subject to separate disposition requirements. Such costs are not appropriately included in decommissioning costs (this principle (in the 10 CFR 50 context) is discussed in Regulatory Guide 1.159 (NRC, 1990), Section 1.4.2, page 1.159-8). Further, the "tails" products from the NEF are not mill tailings, as regulated pursuant to the Uranium Mill Tailings Radiation Control Act, as amended and 10 CFR 40, Appendix A (CFR, 2003j), and are not subject to the financial requirements applicable to mill tailings.

Nevertheless, LES intends to provide for expected tails disposition costs (even assuming ultimate disposal as waste) during the life of the facility. Funds to cover these costs are based on the amount of tails generated and the unit cost for the disposal of depleted UF_6.

It is anticipated that the NEF will generate 132,942 MT of depleted uranium over a nominal 30 year operational period. This estimate is conservative as it assumes continuous production of tails over 30 years of operation. Actual tails production will cease prior to the end of the license term as shown in Figure 10.1-1, NEF – Conceptual Decommissioning Schedule.

Waste processing and disposal costs for UF_6 tails are currently estimated to be $5.50 per kg U or $5,500 per MT U. This unit cost was obtained from four sets of cost estimates for the conversion of DUF_6 to DU_3O_8 and the disposal of DU_3O_8 product, and the transportation of DUF_6 and DU_3O_8. The cost estimates were obtained from analyses of four sources: a 1997 study by the Lawrence Livermore National Laboratory (LLNL) (Elayat, 1997), the Uranium Disposition Services (UDS) contract with the Department of Energy (DOE) of August 29, 2002 (DOE, 2002), information from Urenco, and the costs submitted to the Nuclear Regulatory Commission as part of the Claiborne Enrichment Center (CEC) license application (LES, 1993a) in the 1990s.

The four sets of cost estimates obtained are presented in Table 10.3-1, Summary Of Depleted UF_6 Disposal Costs From Four Sources, below, in 2002 dollars per kg of uranium (kg U). Note that the Claiborne Energy Center cost had a greater uncertainty associated with it. The UDS contract does not allow the component costs for conversion, disposal and transportation to be estimated. The costs in the table indicate that $5.50 per kg U ($2.50 per lb U) is a conservative and, therefore, prudent estimate of total depleted UF_6 disposition cost for the LES NEF. Urenco has reviewed this estimate and, based on its current cost for UBC disposal, finds this figure to be prudent.

In May 1997, the LLNL published UCRL-AR-127650, Cost Analysis Report for the Long-Term Management of Depleted Uranium Hexafluoride (Elayat, 1997). The report was prepared to provide comparative life-cycle cost data for the Department of Energy's (DOE's) Draft 1997 Programmatic Environmental Impact Statement (PEIS) (DOE, 1997) on alternative strategies for management and disposition of DUF_6. The LLNL report is the most comprehensive assessment of DUF_6 disposition costs for alternative disposition strategies available in the public domain.

The technical data on which the LLNL report is based is principally the May 1997 Engineering Analysis Report (UCRL-AR-124080, Volumes 1 and 2) (Dubrin, 1997).

When the LLNL report was prepared in 1997, more than six years ago, the cost estimates in it were based on an inventory of 560,000 MT of DUF_6, or 378,600 MTU after applying the 0.676 mass fraction multiplier. This amount corresponds to an annual throughput rate of 28,000 MT of UF6 or about 19,000 MTU of depleted uranium. The costs in the LLNL report are based on the 20 year life-cycle quantity of 378,600 MTU. The LLNL annual DUF_6 quantities are about 3.6 times the annual production rate of the proposed NEF.

The LLNL cost analyses assumed that the DUF_6 would be converted to DU_3O_8, the DOE's preferred disposal form, using one of two dry process conversion options. The first --- the anhydrous hydrogen fluoride (AHF) option ---- upgrades the hydrogen fluoride (HF) product to anhydrous HF (< 1.0% water). In the second option --- the HF neutralization option --- the hydrofluoric acid would be neutralized with lime to produce calcium fluoride (CaF_2). The LLNL cost analyses assumed that the AHF and CaF_2 conversion products are of sufficient purity that they could be sold for unrestricted use (negligible uranium contamination).

The costs in Table 10.3-1, represent the LLNL-estimated life-cycle capital, operating, and regulatory costs, in 2002 dollars, for conversion of 378,600 MTU over 20 years, of DUF_6 to DU_3O_8 by anhydrous hydrogen fluoride (HF) processing, followed by DU_3O_8 long-term storage disposal in a concrete vault, or in an exhausted underground uranium mine in the western United States, at or below the same cost. An independent new underground mine production cost analysis confirmed that the LLNL concrete vault alternative costs represent an upper bound for under ground mine disposal. The discounted 1996 dollar costs in the LLNL report were undiscounted and escalated to 2002 dollars. The LLNL life-cycle costs in 1996 dollars were converted to per kgU costs and adjusted to 2002 dollars using the Gross Domestic Product (GDP) Implicit Price Deflator (IPD). The escalation adjustment resulted in the 1996 costs being escalated by 11%.

On August 29, 2002, the DOE announced the competitive selection of Uranium Disposition Services, LLC to design, construct, and operate conversion facilities near the DOE enrichment plants at Paducah, Kentucky and Portsmouth, Ohio. UDS will operate these facilities for the first five years, beginning in 2005. The UDS contract runs from August 29, 2002 to August 3, 2010. UDS will also be responsible for maintaining the depleted uranium and product inventories and transporting depleted uranium from Oak Ridge East Tennessee Technology Park (ETTP) to the Portsmouth site for conversion. The DOE-UDS contract scope includes packaging, transporting and disposing of the conversion product DU_3O_8.

UDS is a consortium formed by Framatome ANP Inc., Duratek Federal Services Inc., and Burns and Roe Enterprises Inc. The DOE-estimated value of the cost reimbursement contract is $558 million (DOE Press Release, August 29, 2002) (DOE, 2002). Design, construction and operation of the facilities will be subject to appropriations of funds from Congress. On December 19, 2002, the White House confirmed that funding for both conversion facilities will be included in President Bush's 2004 budget. However, the Office of Management and Budget has not yet indicated how much funding will be allocated. The UDS contract quantities and costs are given in Table 10.3-2, DOE-UDS August 29, 2002, Contract Quantities and Costs.

Urenco is currently contracted with a supplier for DUF_6 to DU_3O_8 conversion. The supplier has been converting DUF_6 to DU_3O_8 on an industrial scale since 1984.

The CEC costs given in Table 10.3-1, are those presented to John Hickey of the NRC in the CEC letter of June 30, 1993 (LES, 1993b) as adjusted for changes in units and escalated to 2002 ($6.74 per kgU). The conversion cost of $4.00 per kg U was provided to CEC by Cogema at that time. It should also be noted that this highest cost estimate is at least 10 years old and was based on the information available at that time. The value of $5.50 per kgU used in the decommissioning cost estimate is 22% above the average of the more recent LLNL and UDS cost estimates, which is $4.49 per kgU {(5.06+3.92)/2}. The LLNL Cost Analysis Report (page 30) states that its cost estimate already includes a 30% contingency in the capital costs of the process and manufacturing facilities, a 20% contingency in the capital costs of the balance of plant; and a minimum of a 30% contingency in the capital costs of process and manufacturing equipment.

Also, the 1997 LLNL cost information is five years older than the more recent 2002 UDS cost information. The value of $5.50 per kgU used in the decommissioning cost estimate for tails disposition is 40% greater than the 2002 UDS-based cost estimate of $3.92 per kgU, which does not include offset credits for HF sales or proceeds from the sale of recycled products.

The costs in Table 10.3-1, indicate that $5.50 is a conservative and, therefore, prudent estimate of total DU disposition cost for the NEF. Urenco has reviewed this estimate and, based on its current cost after tails disposal, finds this figure to be prudent.

In summary, there is already substantial margin between the value of $5.50 per kgU being used by LES in the decommissioning cost estimate and the most recent information (2002 UDS) from which LES derived a cost estimate of $3.92 per kgU.

Based on a computed tails production of 132,942 MTU during a nominal 30 years of operation and a tails processing cost of $5.50/kgU or $5,500 per MTU, the total tails disposition funding requirement is estimated at $731,181,000. This sum will be included as part of the financial assurance for decommissioning (see Table 10.1-14, Total Decommissioning Costs). See Environmental Report Section 4.13.3.1.6, Costs Associated with UF_6 Tails Conversion and Disposal, for additional details.

10.4 REFERENCES

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.38, Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 20.1402, Radiological criteria for unrestricted use, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Part 20.1003, Definitions, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Part 20.2108, Records of waste disposal, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Part 20, Subpart E, Radiological Criteria for License Termination, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Part 20.2002, Method for obtaining approval of proposed disposal procedures, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Part 95, Security Facility Approval and Safeguarding of National Security Information and Restricted Data, 2003.

CFR, 2003h. Title 10, Code of Federal Regulations, Section 40.36, Financial assurance and recordkeeping for decommissioning, 2003.

CFR, 2003i. Title 10, Code of Federal Regulations, Section 70.25, Financial assurance and recordkeeping for decommissioning, 2003.

CFR, 2003j. Title 10, Code of Federal Regulations, Part 40, Appendix A, Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material From Ores Processed Primarily for Their Source Material Content, 2003.

DOE, 1997. Programmatic Environmental Impact Statement for Alternative Strategies for the Long-Term Management and Use of Depleted Uranium Hexafluoride, U.S. Department of Energy, December 1997.

DOE, 2002. Department of Energy Selects Uranium Disposition Services for Uranium Hexafluoride Conversion Plants in Ohio and Kentucky, Department of Energy News Release R-02-179, August 29, 2002.

Dubrin, 1997. “Depleted Uranium Hexafluoride Management Program”, UCRL-AR-124080 Vol. 1 Rev. 2 and Vol. 2, Lawrence Livermore National Laboratory, Dubrin, J.W., et. al., May 1997.

Elayat, 1997. “Cost Analysis Report For the Long-Term Management of Depleted Uranium Hexafluoride”, UCRL-AR-127650, Lawrence Livermore National Laboratory, Elayat, Hatem, J.Zoller, L. Szytel, May 1997.

LES, 1993a. Clairborne Enrichment Center Safety Analysis Report, Section 11.8, Decommissioning, Louisiana Energy Services, 1993.

LES, 1993b. Letter from Peter G. LeRoy, Louisiana Energy Services, to John W.N. Hickey, U.S. Nuclear Regulatory Commission, June 30, 1993.

NRC, 1990. Assuring the Availability of Funds for Decommissioning Nuclear Reactors, Regulatory Guide 1.159, U.S. Nuclear Regulatory Commission, August 1990.

NRC, 1994. Safety Evaluation Report for the Claiborne Enrichment Center, Homer, Louisiana, NUREG-1491, U.S. Nuclear Regulatory Commission, January 1994.

NRC, 2003. Consolidated NMSS Decommissioning Guidance – Financial Assurance, Recordkeeping, and Timeliness, NUREG-1757, Volume 3, U.S. Nuclear Regulatory Commission, September 2003.

TABLES

Table 10.1-1A Number and Dimensions of Facility Components
Page 1 of 1

Separations Modules (Note 1)

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes			
Fume Cupboards			
Lab Benches			
Sinks			
Drains			
Floors			
Walls			
Ceilings			
Ventilation/Ductwork			
Hot Cells			
Equipment/Materials			
Soil Plots			
Storage Tanks			
Storage Areas			
Radwaste Areas			
Scrap Recovery Areas			
Maintenance Shop			
Equipment Decontamination Areas			
Other			

Notes:

1. More than 97% of the decommissioning costs for the facility are attributed to the dismantling, decontamination, processing, and disposal of centrifuges and other equipment in the Separations Building Modules, which are considered classified. Given the classified nature of these buildings, the data presented in these Tables have been structured to meet the applicable NUREG-1757 recommendations, to the extent practicable. However, specific information regarding numbers of components, dimensions of components, and total dimensions, has been intentionally excluded to protect the classified nature of the data.

Table 10.1-1B Number and Dimensions of Facility Components
Page 1 of 1

Decommission Decontamination Facility

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	10	Various sizes of lab and workshop benches ranging from 6.5 to 13 feet long by 2.5 feet wide	(Note 1)
Sinks	6	Standard laboratory sinks and hand wash basins	(Note 1)
Drains	6	Standard laboratory type drains	(Note 1)
Floors	1 Lot (Note 2)	(Note 1)	(Note 1)
Walls	1 Lot (Note 2)	(Note 1)	(Note 1)
Ceilings	1 Lot (Note 2)	(Note 1)	(Note 1)
Ventilation/Ductwork	(Note 3)	Various sizes of ductwork ranging from 3 to 18 inches plus dampers, valves and flexibles	640 feet
Hot Cells	None	NA	NA
Equipment/Materials	20	Various pieces of equipment including citric cleaning tanks, centrifuge cutting machines	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	1 Lot (Note 2)	Various storage tanks	(Note 1)
Storage Areas	1	Storage area for centrifuges and pipe work	(Note 1)
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling and decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table 10.1-1C Number and Dimensions of Facility Components

Page 1 of 1

Technical Services Building

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	18	Standard laboratory fume cupboards, approx 6.5 - 8 feet high x 5 feet wide	(Note 1)
Lab Benches	25	Various sizes of lab and workshop benches ranging from 6.5 – 13 feet long by 2.5 feet wide	(Note 1)
Sinks	12	Standard laboratory sinks and hand wash basins plus larger sinks for laundry	(Note 1)
Drains	12	Standard Laboratory type drains plus larger laundry drain	(Note 1)
Floors	(Note 3)	Floor area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	26,340 ft^2
Walls	(Note 3)	Wall area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	40,074 ft^2
Ceilings	(Note 3)	Ceiling area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	26,340 ft^2
Ventilation/ Ductwork	(Note 3)	Various pieces of equipment including, filter banks, extractor fans, vent stack, dampers and approx 2,034 feet of large and small ductwork	2,034 feet
Hot Cells	None	NA	NA
Equipment/ Materials	57	Various pieces of equipment including, mass spectrometers, washing machines, hydraulic lift tables, cleaning cabinets	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	1	Waste oil storage tank (53 gal)	(Note 1)
Storage Areas	2	Storage area for product removal, dirty pumps	(Note 1)
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table 10.1-1D Number and Dimensions of Facility Components
Page 1 of 1

Gaseous Effluent Vent (GEV) System Throughout Plant

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	None	NA	NA
Sinks	None	NA	NA
Drains	None	NA	NA
Floors	None	NA	NA
Walls	None	NA	NA
Ceilings	None	NA	NA
Ventilation/Ductwork	(Note 3)	Various sizes of ductwork ranging from 3 to 18 inches plus dampers, valves and flexibles	5,656 feet
Hot Cells	None	NA	NA
Equipment/Materials	None	NA	NA
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
RadWaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table 10.1-1E Number and Dimensions of Facility Components
Page 1 of 1

Blending and Sampling

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	None	NA	NA
Sinks	None	NA	NA
Drains	None	NA	NA
Floors	None (Note 4)	NA	NA
Walls	None (Note 4)	NA	NA
Ceilings	None (Note 4)	NA	NA
Ventilation/Ductwork	Covered in GEV System estimate	Covered in GEV System estimate	Covered in GEV System estimate
Hot Cells	None	NA	NA
Equipment/Materials	(Note 3)	Various sizes of pipe-work ranging from DN25 to DN65	2,461 feet
	38 Valves	Various types of valve ranging from 0.6 to 2.5 inches and manual to control	(Note 1)
	12	Various pieces of equipment including hot boxes and traps	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.
4. No floors, walls or ceilings are anticipated needing decontamination.

Table 10.1-1F Number and Dimensions of Facility Components
Page 1 of 1

Centrifuge Test and Post Mortem

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	4	Various sizes of lab and workshop benches ranging from 6.5 – 13 feet long by 2.5 feet wide	(Note 1)
Sinks	2	Standard laboratory sinks and hand wash basins plus larger sinks for laundry	(Note 1)
Drains	2	Standard laboratory type drains plus larger laundry drain	(Note 1)
Floors	None (Note 4)	NA	NA
Walls	None (Note 4)	NA	NA
Ceilings	None (Note 4)	NA	NA
Ventilation/ Ductwork	None	NA	NA
Hot Cells	None	NA	NA
Equipment/ Materials	(Note 3)	Various sizes of pipe-work ranging from DN16 to DN40	164 feet
	56 Valves	Various types of valve ranging from 0.6 to 1.6 inches and manual to control	(Note 1)
	7	Various pieces of equipment including feed take off vessels and traps	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.
4. No floors, walls or ceilings are anticipated needing decontamination.

Table 10.1-2 Planning and Preparation
Page 1 of 1

(Note 1)

Activity	Costs ($000)	Activity Duration (Months)
Project Plan & Schedule	100	4
Site Characterization Plan	200	4
Site Characterization	300	4
Decommissioning Plan	350	6
NRC Review Period	50	12
Site Services Specifications	100	2
Project Procedures	100	4
TOTAL	1,200	(Note 2)

Notes:

1. Deviates from NUREG-1757 because costs are derived from activity durations based on recent commercial decommissioning cost estimates.
2. Some activities will be conducted in parallel to achieve a 24 month time frame.

Table 10.1-3 Decontamination or Dismantling of Radioactive Components (Man Hours)
Page 1 of 1

Other Buildings (Note 1)

Component	Decon Method (Note 4)	Craftsman	Supervision (Note 2)	Project Management (Note 5)	HP&S/Chem (Notes 3 & 5)
Glove Boxes		0	0		
Fume Cupboards		312	62		
Lab Benches		324	64		
Sinks		101	20		
Drains		102	20		
Floors		647	129		
Walls		422	84		
Ceilings		275	55		
Ventilation/Ductwork		8,468	1,693		
Hot Cells		0	0		
Equipment/Materials		1,533	307		
Soil Plots		0	0		
Storage Tanks		14	3		
Storage Areas		110	22		
Radwaste Areas		0	0		
Scrap Recovery Areas		0	0		
Maintenance Shop		0	0		
Equipment Decontamination Areas		0	0		
Other		1,913	382		
TOTAL Hours	--	14,221	2,841	2,430	2,990

Notes:

1. Includes the Decontamination Facility, Technical Services Building, Gaseous Effluent Vent System Throughout Plant, Blending and Sampling, and Centrifuge Test and Post Mortem Facilities.
2. Supervision at 20%.
3. Supply ongoing monitoring and analysis service for dismantling teams.
4. Decontamination method not defined at this time.
5. Total hours allocated based on Urenco decommissioning experience.

Table 10.1-4 Restoration of Contaminated Areas on Facility Grounds
(Work Days)
Page 1 of 1

Activity	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category
Backfill and Restore Site (Note 1)						
TOTAL						

Note:

1. Deviates from NUREG-1757 because cost is based on volume and unit cost associated with removal and disposal of liners and earthen covers of the facility evaporative basins. The cost (see Table 10.1-14) assumes transport and disposal of approximately 33,000 ft^3 of contaminated soil and basin membrane. Other contaminated areas outside of the plant buildings are not expected.

Table 10.1-5 Final Radiation Survey
Page 1 of 1

(Note 1)

Activity	Costs ($000)	Activity Duration (Months)
Prepare Survey Plans and Grid Areas	500	8
Collect Survey Readings and Analyze Data	1,400	16
Final Status Survey Report and NRC Review	300	8
Confirmatory Survey and Report	200	6
Terminate Site License	100	2
TOTAL	2,500	(Note 2)

Notes:

1. Deviates from NUREG-1757 because cost is derived from activity durations based on recent commercial decommissioning cost estimates.
2. Some activities will be conducted in parallel to achieve a 36 month time frame.

Table 10.1-6 Site Stabilization and Long-Term Surveillance
(Work Days)
Page 1 of 1

Activity	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category
(Note 1)	N/A	N/A	N/A	N/A	N/A	N/A

Note:

1. Site stabilization and long-term surveillance will not be required.

Table 10.1-7 Total Work Days by Labor Category
(Based on a 7.5 hr Working Day)
Page 1 of 1

Task	Shift- worker (multi-functional)	Craftsman	Supervision	Project Management	HP&S	Cleaner
Planning and Preparation (see Table 10.1-2)						
Decontamination and/or Dismantling of Radioactive Facility Components	56,067	1,896	6,156	1,478	1,828	2,897
Restoration of Contaminated Areas on Facility Grounds (see Table 10.1-4)						
Final Radiation Survey (see Table 10.1-5)						
Site Stabilization and Long-Term Surveillance (see Table 10.1-6)						

Table 10.1-8 Worker Unit Cost Schedule

Page 1 of 1

Labor Cost Component	Shift- worker (multi-functional)	Craftsman	Supervision	Project Management	HP&S	Cleaner
Salary & Fringe ($/year)	73,006	65,184	96,000	120,000	96,000	73,006
Overhead Rate (%)	excluded	excluded	excluded	excluded	excluded	excluded
Total Cost Per Year ($)	73,006	65,184	96,000	120,000	96,000	73,006
Total Cost Per Work Day ($/day) (Note 1)	342	306	450	563	450	342

Note:

1. Based on 213.33 work days per year at 7.5 hrs per day (1,600 hrs per year)

Table 10.1-9 Total Labor Costs by Major Decommissioning Task
($000)
Page 1 of 1

Task	Shift-worker (multi-functional)	Craftsman	Supervision	Project Management	HP&S	Cleaner
Planning and Preparation (see Table 10.1-2)						
Decontamination and/or Dismantling of Radioactive Facility Components	19,175	579	2,770	832	823	991
Restoration of Contaminated Areas on Facility Grounds (see Table 10.1-4)						
Final Radiation Survey (see Table 10.1-5)						
Site Stabilization and Long-Term Surveillance (see Table 10.1-6)						

Table 10.1-10 Packaging, Shipping and Disposal of Radioactive Wastes
(Excluding Labor Costs)
Page 1 of 1

(a) Waste Disposal Costs (includes packaging & shipping costs)

Waste Type	Disposal Volume (m^3 (ft^3))	Unit Cost ($/$ft^3$)	# of drums	Total Disposal Costs ($000)
Other Buildings :				
Miscellaneous low level waste	83 (2.930)	150	400	440
Separation Modules:				
Solidified Liquid Wastes	432 (15,251)	100	2,159	1,525
Centrifuge Components, Piping and Other Parts	1,036 (36,595)	100	5,180	3,659
Aluminum	3,602 (127,200)	100	NA	12,720
TOTAL	5,153 (181,976)	--	7,739	18,344

(b) Processing Costs

Materials	Disposal Weight (tons)	Unit Cost ($/lb)	Total Disposal Costs ($000)
Aluminum	10,177	0.14	2,860
Other materials	155	2.67	830
TOTAL	10,332	--	3,690

Table 10.1-11 Equipment and Supply Costs
(Excluded Containers)
Page 1 of 1

(a) Equipment

Equipment	Quantity	Unit Cost ($/unit)	Total Cost Equipment ($000)
Separation Building Modules			
Dismantling and decontamination building	45,210 ft^2	1,545	6,490
Special floor and vent system	45,210 ft^2	294	1,240
Plant equipment			
Basic decontamination equipment	lot (Note 1)	600,000	600
Decontamination line equipment	2 units	3,908,850	7,820
Evaporation installation	lot (Note 1)	390,000	390
Radiation and control equipment	lot (Note 1)	410,000	410
Electrical and Instrumentation			
Electrical system	lot (Note 1)	500,000	500
Instrumentation	lot (Note 1)	590,000	590
Design and Engineering			
Building	-	20% (Note 1)	1,550
Plant and equipment	-	15% (Note 1)	1,400
Electrical and Instrumentation	-	25% (Note 1)	270
Other Buildings:			
Dismantling/Cleaning Tools, Equipment and Consumables	lot (Note 1)	100,000	100
TOTAL	--	--	21,360

Note:
1. Allocation based on Urenco decommissioning experience.

(b) Supply

Equipment	Quantity	Unit Cost ($/$ft^3$)	Total Cost Equipment ($000)
Electricity kwh	2,910,344	0.062	180
Gas ft^3	16,900,000	0.004	75
Water ft^3	86,300	0.035	3
Materials	lot (Note 1)		653
TOTAL	--	--	910

Note:
1. Allocation based on Urenco decommissioning experience.

Table 10.1-12 Laboratory Costs
Page 1 of 1

Activity	Quantity	Unit Cost ($)	Total Costs ($000)
Analysis of samples	931	934	870
TOTAL	--	--	870

Table 10.1-13 Period Dependent Costs

Cost Item	Total Cost ($000)
License Fees	(Note 1)
Insurance	(Note 1)
Taxes	(Note 1)
Other	(Note 1)
TOTAL	10,000

Note:

1. Period Dependent Costs include management, insurance, taxes, and other costs for the period beginning with the termination of operations of Separations Building Module 3 and the remaining plant facilities. This assumes $2,000,000 per year for each of the five years at the end of the project. It has been assumed that the period dependent decommissioning costs incurred during concurrent enrichment operations will be funded from operating plant funding and not the decommissioning trust fund.

Table 10.1-14 Total Decommissioning Costs
Page 1 of 2

(Note 7)

Task/Components	Costs ($000)		Total ($000)	Percentage	Notes
	Separations Modules	Other Buildings			
Planning and Preparation (see Table 10.1-2)	1,200	0	1,200	1%	1
Decontamination and Dismantling of Radioactive Facility Components (see Table 10.1-9)	24,060	1,110	25,170	30%	8
Restoration of Contamination Areas on Facility Grounds (see Table 10.1-4)	1,000	0	1,000	1%	2
Final Radiation Survey (see Table 10.1-5)	2,500	0	2,500	3%	3
Site Stabilization and Long-term Surveillance	0	0	0	0%	4
Waste Processing Costs (see Table 10.1-10)	3,690	0	3,690	4%	5
Waste Disposal Costs (see Table 10.1-10)	17,904	440	18,344	22%	6
Equipment Costs (see Table 10.1-11)	21,260	100	21,360	25%	--
Supply Costs (see Table 10.1-11)	910	0	910	1%	--
Laboratory Costs (see Table 10.1-12)	870	0	870	1%	--
Period Dependent Costs (see Table 10.1-13)	10,000	0	10,000	12%	--
SUBTOTAL	83,394	1,650	85,044		--
Contingency (25%)	20,849	413	21,262		--
TOTAL	104,243	2,063	106,306		--
Tails Disposition	0	0	731,181		9
GRAND TOTAL	--	--	837,487		10

Table 10.1-14 Total Decommissioning Costs
Page 2 of 2

Notes:

1. The $1,200 includes planning, site characterization, Decommissioning Plan preparation, and NRC review for the entire plant.
2. Cost provided is for removal and disposal of liners and earthen covers of the facility evaporative basins. The cost assumes transport and disposal of approximately 33,000 ft^3 of contaminated soil and basin membrane at recent commercial rates. Other contaminated areas outside of the plant buildings are not expected.
3. The $2,500 includes the Final Radiation Survey, NRC review, confirmatory surveys and license termination for the entire plant.
4. Site stabilization and long-term surveillance will not be required.
5. Waste processing costs are based on commercial metal melting equipment and unit rates obtained from Urenco experience in Europe.
6. Includes waste packaging and shipping costs. Waste disposal costs for Other Buildings are based on a $150 per cubic foot unit rate which includes packaging, shipping and disposal at Envirocare in Utah.
7. More than 97% of the decommissioning costs for the facility are attributed to the dismantling, decontamination, processing, and disposal of centrifuges and other equipment in the Separations Building Modules, which are considered classified. Given the classified nature of these buildings, the data presented in these Tables have been structured to meet the applicable NUREG-1757 recommendations, to the extent practicable. However, specific information such as numbers of components and unit rates has been intentionally excluded to protect the classified nature of the data. The remaining 3% of the decommissioning costs are for the remaining systems and components in Other Buildings.
8. The $1,110 for Other Buildings includes the decontamination and dismantling of contaminated equipment in the TBS, Blending and Liquid Sampling Area, Centrifuge Test and Post Mortem Facilities, and Gaseous Effluent Vent System.
9. Refer to Section 10.3, for Tails Disposition discussion.
10. Combined total for both decommissioning and tails disposition.

Table 10.3-1 Summary of Depleted UF_6 Disposal Costs from Four Sources
Page 1 of 1

Source	Costs in 2002 Dollars per kgU			
	Conversion	Disposal	Transportation	Total
LLNL (UCRL-AR-127650) (a)	2.64	2.17	0.25	5.06
UDS Contract (b)	(d)	(d)	(d)	3.92
URENCO (e)	(d)	(d)	(d)	(d)
CEC Cost Estimate (c)	4.93	1.47	0.34	6.74

Notes:

(a) 1997 Lawrence Livermore National Laboratory cost estimate study for DOE, discounted costs in 1996 dollars were undiscounted and escalated to 2002 by ERI.
(b) Uranium Disposition Services (UDS) contract with DOE for capital and operating costs for first five years of Depleted UF_6 conversion and Depleted U_3O_8 conversion product disposition.
(c) Based upon Depleted UF_6 and Depleted U_3O_8 disposition costs provided to the NRC during Claiborne Enrichment Center license application in 1993.
(d) Cost component is proprietary or not made available.
(e) The average of the three costs is $5.24/kg U. LES has selected $5.50/kg U as the disposal cost for the National Enrichment Facility. Urenco has reviewed this cost estimate, and based on its current experience with UF_6 disposal, finds this figure to be prudent.

Table 10.3-2 DOE-UDS August 29, 2002, Contract Quantities and Costs
Page 1 of 1

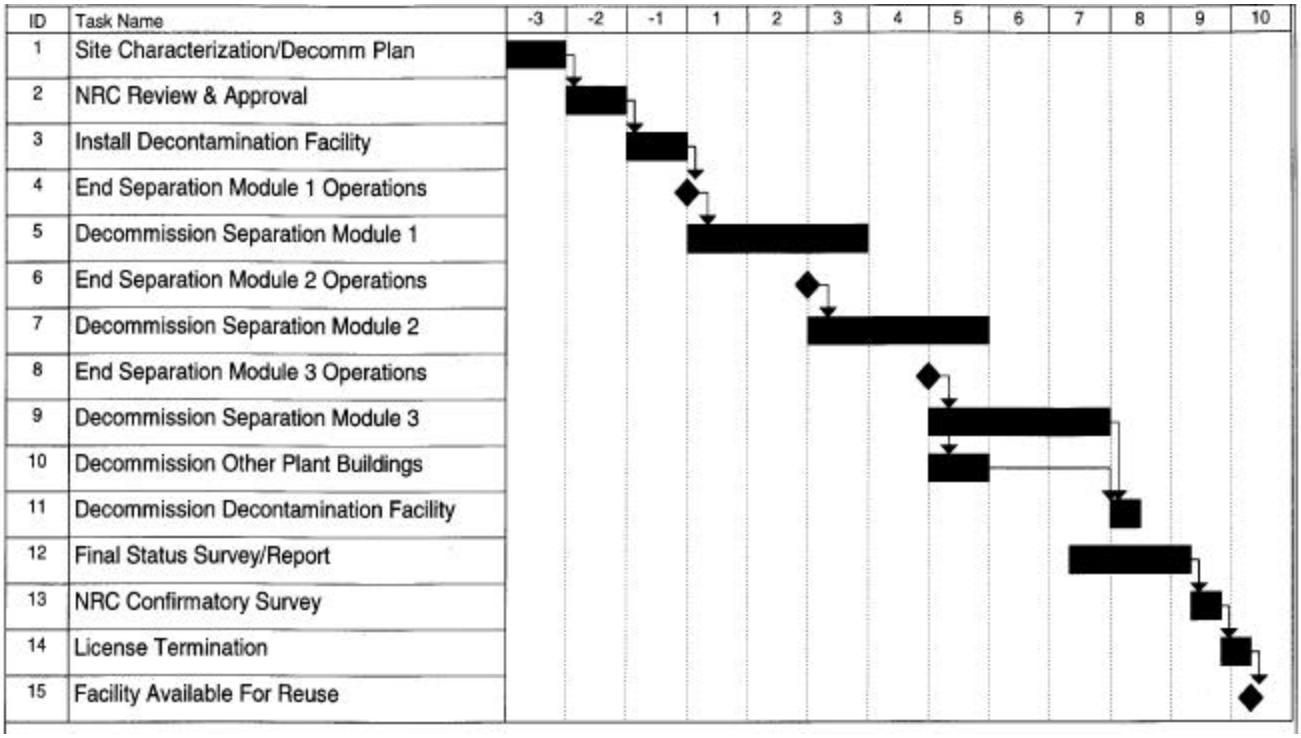
	Target Million kgU	
UDS Conversion and Disposal Quantities:	DUF6 (a)	U (b)
FY 2005 (August-September)	1.050	0.710
FY 2006	27.825	18.800
FY 2007	31.500	21.294
FY 2008	31.500	21.294
FY 2009	31.500	21.294
FY 2010 (October-July)	26.250	17.745
Total:	149.625	101.147
Nominal Conversion Rate (c) and Target Conversion Rate (Million kgU/Yr)		21.3
UDS Contract Workscope Costs: (d)		**Million $**
Design, Permitting, Project Management, etc.		27.99
Construct Paducah Conversion Facility		93.96
Construct Portsmouth Conversion Facility		90.40
Operations for First 5 Years DUF_6 and DU_3O_8 (e)		283.23
Contract Estimated Total Cost w/o Fee		495.58
Contract Estimated Value per DOE PR, August 29, 2003		558.00
Difference Between Cost and Value is the Estimated Fee of 12.6%		62.42
Capital Cost w/o Fee		212.35
Capital Cost with Fee		239.10
First 5 Years Operating Cost with Fee		318.92
Estimated Unit Conversion and Disposal Costs:		
Unit Capital Cost (f)		$0.77/kgU
2005-2010 Unit Operating Costs in 2002 $		$3.15/kgU
Total Estimated Unit Cost		$3.92/kgU

Notes:

(a) As on page B-10 of the UDS contract.
(b) DUF_6 weight multiplied by the uranium atomic mass fraction, 0.676.
(c) Based on page H-34 of the UDS contract.
(d) Workscope costs as on UDS contract pages B-2 and B-3.
(e) Does not include any potential off-set credit for HF sales.
(f) Assumed operation over 25 years, 6% government cost of money, and no taxes.

FIGURES

NATIONAL ENRICHMENT FACILITY - CONCEPTUAL DECOMMISSIONING SCHEDULE



REFERENCE NUMBER
Figure 10.1–1.doc

FIGURE 10.1-1

NATIONAL ENRICHMENT FACILITY – CONCEPTUAL DECOMMISSIONING SCHEDULE

REVISION DATE: DECEMBER 2003

TABLE OF CONTENTS

Page

11.0 MANAGEMENT MEASURES 11.0-1

11.1 CONFIGURATION MANAGEMENT (CM) 11.1-1

11.1.1 Configuration Management Policy 11.1-1
11.1.1.1 Scope of Structures, Systems, and Components 11.1-3
11.1.1.2 Interfaces with Other Management Measures 11.1-3
11.1.1.3 Objectives of Configuration Management 11.1-4
11.1.1.4 Description of Configuration Management Activities 11.1-5
11.1.1.5 Organizational Structure and Staffing Interfaces 11.1-5
11.1.2 Design Requirements 11.1-6
11.1.2.1 Configuration Management Controls on the Design Requirements 11.1-8
11.1.3 Document Control 11.1-9
11.1.4 Change Control 11.1-10
11.1.4.1 Design Phase 11.1-10
11.1.4.2 Construction Phase 11.1-11
11.1.4.3 Operations Phase 11.1-11
11.1.5 Assessments 11.1-12

11.2 MAINTENANCE 11.2-1

11.2.1 Surveillance/Monitoring 11.2-3
11.2.2 Corrective Maintenance 11.2-4
11.2.3 Preventive Maintenance 11.2-4
11.2.4 Functional Testing 11.2-5
11.2.4.1 Objectives 11.2-5
11.2.4.2 Procedure Content 11.2-6
11.2.4.3 Preoperational Testing Program 11.2-7
11.2.4.4 Operational Testing Program 11.2-9

11.3 TRAINING AND QUALIFICATIONS 11.3-1

11.3.1 Organization and Management of the Training Function 11.3-1
11.3.2 Analysis and Identification of Functional Areas Requiring Training 11.3-2
11.3.3 Position Training Requirements 11.3-2
11.3.3.1 General Employee Training 11.3-3
11.3.3.2 Technical Training 11.3-6
11.3.4 Basis and Objectives for Training 11.3-10
11.3.5 Organization of Instruction, Using Lesson Plans and Other Training Guides 11.3-10
11.3.6 Evaluation of Trainee Learning 11.3-11
11.3.7 Conduct of On-the-Job Training 11.3-11
11.3.8 Evaluation of Training Effectiveness 11.3-11
11.3.9 Personnel Qualification 11.3-12
11.3.10 Periodic Personnel Evaluations 11.3-12

TABLE OF CONTENTS

Page

11.4 PROCEDURES DEVELOPMENT AND IMPLEMENTATION 11.4-1

11.4.1 Preparation of Procedures 11.4-3
11.4.2 Administrative Procedures 11.4-4
11.4.3 Procedures.....11.4-5
11.4.4 Changes to Procedures 11.4-7
11.4.5 Distribution of Procedures.....11.4-8

11.5 AUDITS AND ASSESSMENTS.....11.5-1

11.5.1 Activities to be Audited or Assessed 11.5-2
11.5.2 Scheduling of Audits and Assessments 11.5-2
11.5.3 Procedures for Audits and Assessments 11.5-3
11.5.4 Qualifications and Responsibilities for Audits and Assessments 11.5-4

11.6 INCIDENT INVESTIGATIONS AND CORRECTIVE ACTION PROCESS 11.6-1

11.6.1 Incident Investigations.....11.6-1
11.6.2 Corrective Action Process.....11.6-2

11.7 RECORDS MANAGEMENT.....11.7-1

11.8 OTHER QA ELEMENTS 11.8-1

11.9 REFERENCES.....11.9-1

APPENDIX A LES QA PROGRAM DESCRIPTION

11.0 MANAGEMENT MEASURES

Management measures are functions applied to item(s) relied on for safety (IROFS) and any items which may affect the function of IROFS to provide reasonable assurance that the IROFS are available and able to perform their functions when needed. This chapter addresses each of the management measures included in the 10 CFR 70.4 definition of management measures.

Management measures are implemented through a quality assurance (QA) program in accordance with 10 CFR 50, Appendix B (CFR, 2003b). The QA program also provides additional measures for ensuring that the design, construction, operation and decommissioning of IROFS are controlled commensurate with their importance to safety. The Louisiana Energy Services (LES) Quality Assurance Program is described in the LES QA Program Description document included as Appendix A to this chapter. The NRC has evaluated the LES QA Program Description and concluded that the application of QA elements as described in the QA Program Description meets the requirements of 10 CFR 70 (CFR, 2003g) and provides reasonable assurance of protection of public and worker health and safety and the environment (NRC, 2004). The current LES QA Program is also consistent with the QA Program submitted for Nuclear Regulatory Commission (NRC) review in Chapter 10 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC staff evaluated the previous LES QA Program and concluded that the program, when implemented effectively, will meet the requirements of 10 CFR 50, Appendix B (CFR, 1994). The staff concluded in Section 12.3 of NUREG-1491 (NRC, 1994) that the LES QA program was acceptable for the design, construction, start-up, and operation of the enrichment facility. References to the NUREG-1491 (NRC, 1994) sections that document the NRC staff's previous acceptance of these management measures are included in each section as appropriate.

LES maintains full responsibility for assuring that the National Enrichment Facility (NEF) is designed, constructed, tested, and operated in conformance with good engineering practices, applicable regulatory requirements and specified design requirements and in a manner to protect the health and safety of the public. To this end, the LES Quality Assurance Program conforms to the criteria established in 10 CFR 50, Appendix B, Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants (CFR, 2003b). The criteria in 10 CFR 50, Appendix B (CFR, 2003b), are implemented following the commitment to ASME NQA-1-1994, Quality Assurance Program Requirements for Nuclear Facilities (ASME, 1994), as revised by the ASME NQA-1a-1995 Addenda (ASME, 1995).

The QA Program described herein includes design, construction, pre-operational testing, and operation of the facility. This QA Program describes the requirements to be applied for those systems, components, items, and services that have been determined to be QA Level 1 as defined in Appendix A. LES and their contractors implement these requirements through the use of approved procedures. In addition, a quality assurance program as described in Appendix A is applied to certain other systems, components, items, and services which are not QA Level 1. The information provided in this chapter, the corresponding regulatory requirement, and the section of NUREG-1520 (NRC, 2002), Chapter 11 in which the NRC acceptance criteria are presented is summarized below.

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 11 Reference
Section 11.1 Configuration Management	70.62(d) & 70.72	11.4.3.1
Section 11.2 Maintenance	70.62(d)	11.4.3.2
Section 11.3 Training and Qualifications	70.62(d) & 10CFR19	11.4.3.3
Section 11.4 Procedures Development and Implementation	70.62(d) & 70.22(a)(8)	11.4.3.4
Section 11.5 Audits and Assessments	70.62(d)	11.4.3.5
Section 11.6 Incident Investigations and Corrective Action Process	70.74(a)&(b) 70.62(a)(3)	11.4.3.6
Section 11.7 Records Management	70.62(a)(2)&(3) 70.62(d)	11.4.3.7
Section 11.8 Other QA Elements	70.62(d)	11.4.3.8
Appendix A: LES QA Program Description	70.62(d)	11.4.3.8

11.1 CONFIGURATION MANAGEMENT (CM)

This section describes the configuration management program for the NEF. Configuration management for the NEF is implemented through requirements of the QA Program and associated procedures.

The LES President is the executive responsible for quality assurance and is the highest level of management responsible for LES's QA policies, goals, and objectives. The President receives policy direction from the LES Management Committee. The LES organization during the design, construction and operation phases, including QA, is presented in Chapter 2, Organization and Administration.

11.1.1 Configuration Management Policy

Configuration management is provided throughout facility design, construction, testing, and operation. Configuration management provides the means to establish and maintain a technical baseline for the facility based on clearly defined requirements. During design and construction, the Engineering and Contracts Manager has responsibility for configuration management through the design control process. Selected documentation, including the integrated safety analysis (ISA), is controlled under the configuration management system in accordance with procedures associated with design control, document control, and records management. Design changes undergo formal review, including interdisciplinary reviews as appropriate, in accordance with these procedures. This interdisciplinary review includes as a minimum the review for ISA impacts.

Configuration management provides the means to establish and maintain the essential features of the design basis of IROFS, including the ISA. As the project progresses from design and construction to operation, configuration management is maintained by the Technical Services organization as the overall focus of activities changes. Procedures will define the turnover process and responsibilities since construction will continue on new work modules during operations.

During the design phase of the project, configuration management is based on the design control provisions and associated procedural controls over design documents to establish and maintain the technical baseline. Design documents, including the ISA, that provide design input, design analysis, or design results specifically for IROFS are identified with the appropriate QA level. These design documents undergo interdisciplinary review during the initial issue and during each subsequent revision. During the construction phase of the project, changes to drawings and specifications issued for construction, procurement, or fabrication are systematically reviewed and verified, evaluated for impact, including impact to the ISA, and approved prior to implementation. Proper implementation is verified and reflected in the design basis documentation.

In order to provide for the continued safe and reliable operation of the facility structures, systems and components, measures are implemented to ensure that the quality of these structures, systems and components is not compromised by planned changes (modifications). After issuance of the Operating License, the Plant Manager is responsible for the design of and modifications to facility structures, systems or components. The design and implementation of modifications are 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 modifications during the operations phase are contained in procedures that are approved, including revisions, by the Technical Services Manager. The modification procedure contains the following items necessary to ensure quality in the modification program:

- The technical and quality requirements which shall be met to implement a modification
- The requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The facility modification procedure shall be written to ensure that policies are formulated and maintained to satisfy the LES QA Program, as applicable.

Each change to the facility or to activities of personnel shall have an evaluation performed in accordance with the requirements of 10 CFR 70.72 (CFR, 2003e), as applicable. Each modification shall also be evaluated for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents.

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures), that involves or could affect uranium on site, a Nuclear Criticality Safety Evaluation (NCSE) shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with applicable margin for safety) under both normal and credible abnormal conditions.

Each modification is also evaluated and documented for radiation exposure to minimize worker exposures in keeping with the facility as low as reasonably achievable (ALARA) program, criticality and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include, but are not limited to the review of:

- Modification cost
- Lessons learned from similar completed modifications
- QA requirements
- Potential operability or maintainability concerns
- Constructability concerns
- Post-modification testing requirements
- Environmental considerations
- Human factors
- Integrated safety analysis.

After completion of a modification to a structure, system, or component, the modification Project Manager, or designee, shall ensure that all applicable 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, e.g., the revised process description, checklists for operation and flowsheets are made available to operations and maintenance departments prior to the start-up of the modified system. Appropriate training on the modification is completed before a system is placed in operation. A formal notice of a modification being completed is distributed to all appropriate managers. As-built drawings incorporating the modification are completed in accordance with the design control procedures. These records shall be identifiable and shall be retained in accordance with the records management procedures.

11.1.1.1 Scope of Structures, Systems, and Components

The scope of Structures, Systems, and Components (SSC) under configuration management includes all IROFS identified by the integrated safety analysis of the design bases and any items which may affect the function of the IROFS. The list of IROFS is provided in Chapter 3, Integrated Safety Analysis Summary. Design documents subject to configuration management include calculations, safety analyses, design criteria, engineering drawings, system descriptions, technical documents, and specifications that establish design requirements for IROFS. During the design phase, these design documents are maintained under configuration management when initially approved.

The scope of documents included in the configuration management program expands throughout the design process. As drawings and specification sections related to IROFS or items affecting the functions of IROFS are prepared and issued for procurement, fabrication, or construction, these documents are included in configuration management.

During construction, initial startup, and operations, the scope of documents under configuration management similarly expands to include, as appropriate: vendor data; test data; inspection data; initial startup, test, operating and administrative procedures as applicable to IROFS and nonconformance reports. These documents include documentation related to IROFS that is generated through functional interfaces with QA, maintenance, and training and qualifications of personnel. Configuration management procedures will provide for evaluation, implementation, and tracking of changes to IROFS, and processes, equipment, computer programs, and activities of personnel that impact IROFS.

11.1.1.2 Interfaces with Other Management Measures

Configuration management is implemented through or otherwise related to other management measures. Key interfaces and relationships to other management measures are described below:

- **Quality Assurance -** The QA program establishes the framework for configuration management and other management measures for IROFS and items that affect the function of the IROFS.

- **Records Management -** Records associated with IROFS and items affecting IROFS are generated and processed in accordance with the applicable requirements of the QA Program and provide evidence of the conduct of activities associated with the configuration management of those IROFS.

- **Maintenance –** Maintenance requirements are established as part of the design basis, which is controlled under configuration management. Maintenance records for IROFS and items affecting IROFS provide evidence of compliance with preventative and corrective maintenance schedules.

- **Training and Qualifications -** Training and qualification are controlled in accordance with the applicable provisions of the QA Program. Personnel qualifications and/or training to specific processes and procedures are management measures that support the safe operation, maintenance, or testing of IROFS. Also, work activities that are themselves IROFS, (i.e., administrative controls) are proceduralized, and personnel are trained and qualified to these procedures. Training and qualification requirements and documentation of training may be considered part of the design basis controlled under configuration management. Reference Sections 11.3.2, Analysis and Identification of Functional Areas Requiring Training, and 11.3.3, Position Training Requirements, for interfaces with configuration management.

- **Incident Investigation/Audits and Assessments -** Audits, assessments, and incident investigations are described in Sections 11.5, Audits and Assessments, and 11.6, Incident Investigations and Corrective Action Process. Corrective actions identified as a result of these management measures may result in changes to design features, administrative controls, or other management measures (e.g., operating procedures). The Corrective Action Program (CAP) is described in Section 11.6, "Incident Investigations and Corrective Action Process." Changes are evaluated under the provisions of configuration management through the QA Program and procedures. Periodic assessments of the configuration management program are also conducted in accordance with the audit and assessment program described in Section 11.5.

- **Procedures -** Operating, administrative, maintenance, and emergency procedures are used to conduct various operations associated with IROFS and items affecting IROFS and will be reviewed for potential impacts to the design basis. Also, work activities that are themselves IROFS, (i.e., administrative controls) are contained in procedures.

11.1.1.3 Objectives of Configuration Management

The objectives of configuration management are to ensure design and operation within the design basis of IROFS by: identifying and controlling preparation and review of documentation associated with IROFS; controlling changes to IROFS; and maintaining the physical configuration of the facility consistent with the approved design.

The Urenco technology transfer documentation provides the enrichment plant design, and identifies those safety trips and features credited in the European safety analyses. The ISA of the design bases determines the IROFS and establishes the safety function(s) associated with each IROFS. Configuration control is accomplished during design through the use of

procedures for controlling design, including preparation, review (including interdisciplinary review), design verification where appropriate, approval, and release and distribution for use. Engineering documents will be assessed for QA level classification. Changes to the approved design are subject to a review to ensure consistency with the design bases of IROFS. Configuration verification is also accomplished through design verification, which ensures that design documents are consistent and that design requirements for IROFS are met. During construction and testing, this verification also extends to verification that as-built configurations are consistent with the design, and that testing that is specified to demonstrate performance of IROFS is accomplished successfully. Periodic audits and assessments of the configuration management program and of the design confirm that the system meets its goals and that the design is consistent with the design bases. The corrective action process occurs in accordance with the LES QA Program and associated procedures in the event problems are identified. Prompt corrective actions are developed as a result of incident investigations or in response to audit or assessment results.

11.1.1.4 Description of Configuration Management Activities

Configuration management includes those activities conducted under design control provisions for ensuring that design and construction documentation is prepared, reviewed, and approved in accordance with a systematic process. This process includes interdisciplinary reviews appropriate to ensure consistency between the design and the design bases of IROFS. During construction, it also includes those activities that ensure that construction is consistent with design documents. Finally, it includes activities that provide for operation of the IROFS in accordance with the limits and constraints established in the ISA, and that provide for control of changes to the facility in accordance with 10 CFR 70.72 (CFR, 2003e).

Configuration management also includes records to demonstrate that personnel conducting activities that are relied on for safety or that are associated with IROFS are appropriately qualified and trained to conduct that work.

Implementing documents are controlled within the document control system. These documents support configuration management by ensuring that only reviewed and approved procedures, specifications and drawings are used for procurement, construction, installation, testing, operation, and maintenance of IROFS, as appropriate.

11.1.1.5 Organizational Structure and Staffing Interfaces

The configuration management program is administered by the Engineering and Contracts organization during design and construction. Engineering includes engineering disciplines with responsible lead engineers in charge of each discipline, under the direction of design managers or project managers who report to the Engineering and Contracts Manager. The lead discipline engineers have primary technical responsibility for the work performed by their disciplines, and are responsible for the conduct of interdisciplinary reviews as discussed previously in this section. Reviews are also conducted, as appropriate, by construction management, operations, QA, and procurement personnel. The design control process also interfaces with the document control and records management process via procedures.

The various LES departments and contractors of LES perform quality-related activities. The primary LES contractors are responsible for development of their respective QA Programs,

which shall be consistent with the requirements of the LES QA Program for those activities determined to be within the scope of the LES QA Program. The interfaces between contractors and LES or among contractors shall be documented. LES and contracted personnel have the responsibility to identify quality problems. If a member of another area disagrees, that individual is instructed to take the matter to appropriate management. The disagreement may either be resolved at this level or at any level up to and including the LES President.

11.1.2 Design Requirements

Design requirements and associated design bases are established and maintained by the Engineering and Contracts organization during design and construction and by the Technical Services organization during operations. The configuration management controls on design requirements and the integrated safety analysis of the design bases are described previously in this section. Design requirements are documented in a design requirements document that provides for a hierarchical distribution of these requirements through basis of design documents. The design requirements document and basis of design documents are controlled under the design control provisions of the configuration management program as described above, and are subject to the same change control as analyses, specifications, and drawings. Computer codes used in the design of IROFS are also subject to these design control measures, with additional requirements as appropriate for software control, verification, and validation.

IROFS, any items that affect the function of the IROFS, and, in general, items required to satisfy regulatory requirements are designated as QA Level 1. The associated design documents are subject to interdisciplinary reviews and design verification. Analyses constituting the integrated safety analysis of the design bases are subject to the same requirements. Changes to the design are evaluated to ensure consistency with the design bases.

IROFS are listed in the design requirements document. This list will be augmented and maintained current as appropriate as IROFS are identified in more detail during detailed design.

A qualified individual who specifies and includes the appropriate codes, standards, and licensing commitments within the design documents prepares each design document, such as a calculation, specification, procedure, or drawing. This individual also notes any deviations or changes from such standards within the design documentation package. Each design document is then checked by another individual qualified in the same discipline and is reviewed for concept and conformity with the design inputs. These design inputs are in sufficient detail to permit verification of the document. The manager having overall responsibility for the design function approves the document. The Engineering Manager documents the entire review process in accordance with approved procedures. These procedures include provisions to assure that appropriate quality standards are specified in design documents, including quantitative or qualitative acceptance criteria. The QA Director conducts audits on the design control process using independent technically qualified individuals to augment the QA audit team.

During the check and review, emphasis is placed on assuring conformance with applicable codes, standards and license application design commitments. The individuals in engineering assigned to perform the check and review of a document have full and independent authority to withhold approval until questions concerning the work have been resolved. Design reviews, alternative calculations, or qualification testing accomplishes verification of design. The bases

for a design, such as analytical models, theories, examples, tables, codes and computer programs must be referenced in the design document and their application verified during check and review. Model tests, when required to prove the adequacy of a concept or a design, are reviewed and approved by the responsible qualified individual. Testing used for design verification shall demonstrate adequacy of performance under conditions that simulate the most adverse design conditions. The tests used for design verification must meet all the design requirements.

Qualified individuals other than those who performed the design but may be from the same organization perform design verification. Verification may be performed by the supervisor of the individual performing the design, provided this need is documented, approved in advance by the supervisor's management, and the supervisor did not specify a singular design approach or rule out certain design considerations, and did not establish the design inputs used in the design or, provided the supervisor is the only individual in the organization competent to perform the verification. The verification by a supervisor of their own design constraints, design input, or design work would only occur in rare instances. This would occur only when the supervisor is the only individual in the organization competent to perform the verification. These instances are authorized and documented in writing on a case-by-case basis.

Independent design verification shall be accomplished before the design document (or information contained therein) is used by other organizations for design work or to support other activities such as procurement, construction, or installation. When this is not practical due to time constraints, the unverified portion of the document is identified and controlled. In all cases, the design verification shall be completed before relying on the item to perform its function or installation becomes irreversible. Any changes to the design and procurement documents, including field changes, must be reviewed, checked and approved commensurate with the original approval requirements.

After design documents have been properly prepared, checked, reviewed, and approved by the appropriate parties, the responsible engineer sends the document to document control for distribution. When required, each recipient of a design document verifies receipt of such document to the document control center.

The document control center, after verification of distribution to a recipient, maintains the required documentation in its files.

When deficiencies are identified which affect the design of IROFS, such deficiencies are documented and resolved in accordance with approved CAP procedures. In accordance with the CAP the report is forwarded for appropriate review to the responsible manager, who coordinates further review of the problem and revises all design documents affected by the deficiency as necessary. Where required, the responsible manager forwards the report to the engineers in other areas, who coordinate necessary revisions to their affected documents

Design interfaces are maintained by communication among the principals. Methods by which this is accomplished include the following:

A. Design documents are reviewed by the responsible engineer or authorized representative. As appropriate, subsequent review or waiver of review by the other area engineers is documented.

B. Project review meetings are scheduled and held to coordinate design, procurement, construction and pre-operational testing of the facility. These meetings provide a primary working interface among the principal organizations.

C. Reports of nonconformances are transmitted and controlled by procedures. As required by the nonconformance procedure, the QA Director/Manager or designee approves resolution of nonconformances.

During the operational phase, measures are provided to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties.

11.1.2.1 Configuration Management Controls on the Design Requirements

Configuration control is accomplished during design through the use of procedures for controlling design, including preparation, review (including interdisciplinary review and preparation of NCSEs as applicable), and design verification where appropriate, approval, and release and distribution for use. Engineering documents are assessed for QA level classification. Changes to the approved design also are subject to a review to ensure consistency with the design bases of IROFS.

Configuration verification is also accomplished through design verification, which ensures that design documents are consistent and that design requirements for IROFS are met. During construction and testing, this verification also extends to verification that as-built configurations are consistent with the design, and that testing that is specified to demonstrate performance of IROFS is accomplished successfully.

The QA Program requires procedures that specify that work performed shall be accomplished in accordance with the requirements and guidelines imposed by applicable specifications, drawings, codes, standards, regulations, quality assurance criteria and site characteristics.

Acceptance criteria established by the designer are incorporated in the instructions, procedures and drawings used to perform the work. Documentation is maintained, including test results, and inspection records, demonstrating that the work has been properly performed. Procedures also provide for review, audit, approval and documentation of activities affecting the quality of items to ensure that applicable criteria have been met.

Maintenance, modification, and inspection procedures are reviewed by qualified personnel knowledgeable in the quality assurance disciplines to determine:

A. The need for inspection, identification of inspection personnel, and documentation of inspection result

B. That the necessary inspection requirements, methods, and acceptance criteria have been identified.

Facility procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure on a frequency determined by the age and use of the procedure to determine if changes are necessary or desirable. Procedures are also reviewed to ensure procedures are maintained up-to-date with facility configuration. These reviews are intended to ensure that any modifications to facility systems, structures or components are reflected in current maintenance, operations and other facility procedures.

11.1.3 Document Control

Procedures are established which control the preparation and issuance of documents such as manuals, instructions, drawings, procedures, specifications, procurement documents and supplier-supplied documents, including any changes thereto. Measures are established to ensure documents, including revisions, are adequately reviewed, approved, and released for use by authorized personnel.

Document control procedures require documents to be transmitted and received in a timely manner at appropriate locations including the location where the prescribed activity is to be performed. Controlled copies of these documents and their revisions are distributed to and used by the persons performing the activity.

Superseded documents are destroyed or are retained only when they have been properly labeled. Indexes of current documents are maintained and controlled.

Document control is implemented in accordance with procedures. An electronic document management system is used both to file project records and to make available the latest revision (i.e., the controlled copy) of design documents. The system provides an “official” copy of the current document, and personnel are trained to use this system to retrieve controlled documents. The system is capable of generating indices of controlled documents, which are uniquely numbered (including revision number). Controlled documents are maintained until cancelled or superseded, and cancelled or superseded documents are maintained as a record, currently for the life of the project or termination of the license, whichever occurs later. Hard-copy distribution of controlled documents is provided when needed in accordance with applicable procedures (e.g., when the electronic document management system is not available).

A part of the configuration management program, the document control and records management procedures, as appropriate, capture the following documents:

- Design requirements, through the controlled copy of the design requirements document
- The design bases, through the controlled copy of the basis of design documents
- The integrated safety analysis of the design bases of IROFS, through the controlled copies of supporting analyses
- Nuclear Criticality Safety Evaluations
- As-built drawings
- Specifications
- All procedures that are IROFS
- Procedures involving training
- QA

- Maintenance
- Audit and assessment reports
- Emergency operating procedures
- Emergency response plans
- System modification documents
- Assessment reports
- Engineering documents including analyses, specifications, technical reports, and drawings.

These items are documented in approved procedures.

11.1.4 Change Control

Procedures control changes to the technical baseline. The process includes an appropriate level of technical, management, and safety review and approval prior to implementation. During the design phase of the project, the method of controlling changes is the design control process described in the QA Program. This process includes the conduct of interdisciplinary reviews that constitute a primary mechanism for ensuring consistency of the design with the design bases. During both construction and operation, appropriate reviews to ensure consistency with the design bases of IROFS and the ISA, respectively, will similarly ensure that the design is constructed and operated/modified within the limits of the design basis. Additional details are provided below.

11.1.4.1 Design Phase

Changes to the design include a systematic review of the design bases for consistency. In the event of changes to reflect design or operational changes from the established design bases, both the integrated safety analysis and other documents affected by design bases of IROFS including the design requirements document and basis of design documents, as applicable are properly modified, reviewed, and approved prior to implementation. Approved changes are made available to personnel through the document control function discussed previously in this section.

During design (i.e., prior to issuance of the NEF Materials License), the method of ensuring consistency between documents, including consistency between design changes and the safety assessment, is the interdisciplinary review process. The interdisciplinary reviews ensure design changes either (1) do not impact the ISA, (2) are accounted for in subsequent changes to the ISA, or (3) are not approved or implemented. Prior to issuance of the License, LES will notify the NRC of potential changes that reduce the level of commitments or margin of safety in the design bases of IROFS.

11.1.4.2 Construction Phase

When the project enters the construction phase, changes to documents issued for construction, fabrication, and procurement will be documented, reviewed, approved, and posted against each affected design document. Vendor drawings and data also undergo an interdisciplinary review to ensure compliance with procurement specifications and drawings, and to incorporate interface requirements into facility documents.

During construction, design changes will continue to be evaluated against the approved design bases. Changes are expected to the design as detailed design progresses and construction begins. A systematic process consistent with the process described above will be used to evaluate changes in the design against the design bases of IROFS and the ISA. Upon issuance of the NEF Materials License, the configuration change process will fully implement the provisions of 10 CFR 70.72 (CFR, 2003e), including reporting of changes made without prior NRC approval as required by 10 CFR 70.72(d)(2) and (3). Any change that requires Commission approval, will be submitted as a license amendment request as required by 10 CFR 70.72(d)(1) and the change will not be implemented without prior NRC approval.

11.1.4.3 Operations Phase

During the operations phase, changes to design will also be documented, reviewed, and approved prior to implementation. LES will implement a change process that fully implements the provisions of 10 CFR 70.72 (CFR, 2003e). Measures are provided to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties.

In order to provide for the continued safe and reliable operation of the facility structures, systems and components, measures are implemented to ensure that the quality of these structures, systems and components is not compromised by planned changes (modifications). After issuance of the Operating License, the Plant Manager is responsible for the design of and modifications to facility structures, systems or components. The design and implementation of modifications are performed in a manner so as to assure quality is maintained in the remainder of the system that is being modified, or as dictated by applicable regulations.

The administrative instructions for modifications are contained in a facility administrative procedure that is approved, including revisions, by the Technical Services Manager with concurrence of the Quality Assurance Manager. The modification procedure contains the following items necessary to ensure quality in the modification program:

- The requirements that shall be met to implement a modification
- The requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The facility modification procedure shall be written to ensure that policies are formulated and maintained to satisfy the quality assurance requirements specified in the LES QA Program, as applicable.

Each change to the facility or to activities of personnel shall have an evaluation performed in accordance with the requirements of 10 CFR 70.72 (CFR, 2003e), as applicable. Each modification shall also be evaluated for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents.

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures) that involves or could affect uranium on site, an NCSE shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with applicable margin for safety) under both normal and credible abnormal conditions.

Each modification is also 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 are not limited to the review of:

- Modification cost
- Lessons learned from similar completed modifications
- QA aspects
- Potential operability or maintainability concerns
- Constructability concerns
- Post-modification testing requirements
- Environmental considerations
- Human factors.

After completion of a modification to a structure, system, or component, the modification Project Manager, or designee, shall ensure that all applicable 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, e.g., the revised process description, checklists for operation and flowsheets are made available to operations and maintenance departments once the modified system becomes “operational.” Appropriate training on the modification is completed before a system is placed in operation. A formal notice of a modification being completed is distributed to all appropriate managers. As-built drawings incorporating the modification are completed promptly. These records shall be identifiable and shall be retained for the duration of the facility license.

11.1.5 Assessments

Periodic assessments of the configuration management program are conducted to determine the system's effectiveness and to correct deficiencies. These assessments include review of the adequacy of documentation and system walk downs of the as-built facility. Such audits and assessments are conducted and documented in accordance with procedures and scheduled as discussed in Appendix A, Section 18, “Audit Schedules.”

Periodic audits and assessments of the configuration management program and of the design confirm that the system meets its goals and that the design is consistent with the design bases. Incident investigations occur in accordance with the QA Program and associated CAP procedures in the event problems are encountered. Prompt corrective actions are developed as a result of incident investigations or in response to adverse audit/assessment results, in accordance with CAP procedures.

11.2 MAINTENANCE

This section outlines the maintenance and functional testing programs to be implemented for the operations phase of the facility. Preventive maintenance activities, surveillance, and performance trending provide reasonable and continuing assurance that IROFS will be available and reliable to perform their safety functions.

The purpose of planned and scheduled maintenance for IROFS is to ensure that the equipment and controls are kept in a condition of readiness to perform the planned and designed functions when required. Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance function is administratively closely coupled to operations. The Maintenance organization plans, schedules, tracks, and maintains records for maintenance activities.

In order to provide for the continued safe and reliable operation of the facility structures, systems and components, measures are implemented to ensure that the quality of these structures, systems and components is not compromised by planned changes (modifications) or maintenance activities. After issuance of the Operating License, the Plant Manager is responsible for the design of and modifications to facility structures, systems or components and all maintenance activities. The design and implementation of modifications are 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 modifications are contained in a facility administrative procedure that is approved, including revisions, by the Technical Services Manager with concurrence of the Quality Assurance Manager. The modification procedure contains 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 facility modification procedure shall be written to ensure that policies are formulated and maintained to satisfy the quality assurance standards specified in the LES QA Program, as applicable.

Listed below are methods or practices that will be applied to the corrective, preventive, and functional-test maintenance elements. LES will prepare written procedures for performance of these methods and practices. These methods and practices include, as applicable:

Authorized work instructions with detailed steps and a reminder of the importance of the IROFS identified in the ISA Summary:

- Parts lists
- As-built or redlined drawings
- A notification step to the Operations function before conducting repairs and removing an IROFS from service

- Radiation Work Permits
- Replacement with like-kind parts and the control of new or replacement parts to ensure compliance with 10 CFR 21 (CFR, 2003a)
- Compensatory measures while performing work on IROFS
- Procedural control of removal of components from service for maintenance and for return to service
- Ensuring safe operations during the removal of IROFS from service
- Notification to Operations personnel that repairs have been completed.

Written procedures for the performance of maintenance activities include the steps listed above. The details of maintenance procedure acceptance criteria, reviews, and approval are provided in Section 11.4, Procedures Development and Implementation.

As applicable, contractors that work on or near IROFS identified in the ISA Summary will be required by LES to follow the same maintenance procedures described for the corrective, preventive, functional testing, or surveillance/monitoring activities listed above for the maintenance function.

Maintenance procedures involving IROFS commit to the topics listed below for corrective and preventive maintenance, functional testing after maintenance, and surveillance/monitoring maintenance activities:

- Pre-maintenance activities require reviews of the work to be performed, including procedure reviews for accuracy and completeness.
- New procedures or work activities that involve or could affect uranium on site require preparation and approval of an NCSE.
- Steps that require notification of all affected parties (operators and appropriate managers) before performing work and on completion of maintenance work. The discussion includes potential degradation of IROFS during the planned maintenance.
- Control of work by comprehensive procedures to be followed by maintenance technicians. Maintenance procedures are reviewed by the various safety disciplines, including criticality, fire, radiation, industrial, and chemical process safety. The procedures describe, as a minimum, the following:
    - Qualifications of personnel authorized to perform the maintenance, functional testing or surveillance/monitoring
    - Controls on and specification of any replacement components or materials to be used (this will be controlled by Configuration Management, to ensure like-kind replacement and adherence to 10 CFR 21 (CFR, 2003a))
    - Post-maintenance testing to verify operability of the equipment
    - Tracking and records management of maintenance activities

- Safe work practices (e. g., lockout/tag out, confined space entry, moderation control or exclusion area, radiation or hot work permits, and criticality, fire, chemical, and environmental issues).

Maintenance activities generally fall into the following categories:

- Surveillance/monitoring
- Corrective maintenance
- Preventive maintenance
- Functional testing.

These maintenance categories are discussed in the following sections.

11.2.1 Surveillance/Monitoring

Surveillance/monitoring is utilized to detect degradation and adverse trends of IROFS so that action may be taken prior to component failure. The monitored parameters are selected based upon their ability to detect the predominate failure modes of the critical components. Data sources include; surveillance, periodic and diagnostic test results, plant computer information, operator rounds, walk downs, as-found conditions, failure trending, and predictive maintenance. Surveillance/monitoring and reporting is required for SSC that are identified as IROFS and any SSC and administrative controls that could impact the functions of an IROFS.

Plant performance criteria are established to monitor plant performance and to monitor IROFS functions and component parameters. These criteria are established using Urenco industry experience, operating data, surveillance data, and plant equipment operating experience. These criteria ensure the reliability and availability of IROFS. The performance criteria are also used to demonstrate that the performance or condition of an IROFS is being effectively controlled through appropriate predictive and repetitive maintenance strategies so that IROFS remain capable of performing their intended function.

Surveillance of IROFS is performed at specified intervals. The purpose of the surveillance program is to measure the degree to which IROFS meet performance specifications. The results of surveillances are trended, and when the trend indicates potential IROFS performance degradation, preventive maintenance frequencies are adjusted or other appropriate corrective action is taken.

Incident investigations may identify root causes of failures that are related to the type or frequency of maintenance. The lessons learned from such investigations are factored into the surveillance/monitoring and preventive maintenance programs as appropriate.

Maintenance procedures prescribe compensatory measures, if appropriate, for surveillance tests of IROFS that can be performed only while equipment is out of service.

Records showing the current surveillance schedule, performance criteria, and test results for all IROFS will be maintained in accordance with the Record Management System.

Results of surveillance/monitoring activities related to IROFS via the configuration management program will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

11.2.2 Corrective Maintenance

Corrective maintenance involves repair or replacement of equipment that has unexpectedly degraded or failed. Corrective maintenance of IROFS restores the equipment to acceptable performance through a planned, systematic, controlled, and documented approach for the repair and replacement activities.

Following any corrective maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the IROFS, if necessary, is performed to ensure the IROFS performs its intended safety function.

The CAP requires facility personnel to determine the cause of conditions adverse to quality and promptly act to correct these conditions.

Results of corrective maintenance activities related to IROFS via the configuration management program will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

11.2.3 Preventive Maintenance

Preventive maintenance (PM) includes preplanned and scheduled periodic refurbishment, partial or complete overhaul, or replacement of IROFS, if necessary, to ensure their continued safety function. Planning for preventive maintenance includes consideration of results of surveillance and monitoring, including failure history. PM also includes instrument calibration and testing.

The PM program procedures and calibration standards (traceable to the national standards system or to nationally accepted calibration techniques, as appropriate) enable the facility personnel to calibrate equipment and monitoring devices important to plant safety and safeguards. Testing performed on IROFS that are not redundant will provide for compensatory measures to be put into place to ensure that the IROFS function is performed until it is put back into service.

Urenco's extensive experience in the industry (30 years) is used to determine initial PM frequencies and procedures. In determining the frequency of PM, consideration is given to appropriately balancing the objective of preventing failures through maintenance against the objective of minimizing unavailability of IROFS because of PM. In addition, feedback from PM and corrective maintenance and the results of incident investigations and identified root causes are used, as appropriate, to modify the frequency or scope of PM. The rationale for deviations from industry standards or vendor recommendations for PM shall be documented.

After conducting preventive maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the SSC, if necessary, is performed to ensure the IROFS performs its intended safety function. Functional testing is described in detail in Section 11.2.4, Functional Testing.

All records pertaining to preventive maintenance will be maintained in accordance with the Records Management System.

Results of preventive maintenance activities related to IROFS via the configuration management system will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

11.2.4 Functional Testing

Functional testing of IROFS is performed as appropriate following initial installation, as part of periodic surveillance testing, and after corrective or preventive maintenance or calibration to ensure that the item is capable of performing its safety function when required.

The overall testing program is broken into the two major testing programs and within each testing program are two testing categories:

A. Preoperational Testing Program

   1. Functional Testing

   2. Initial Startup Testing.

B. Operational Testing Program

   1. Periodic Testing

   2. Special Testing.

Results of surveillance/monitoring activities related to IROFS via the configuration management program will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

11.2.4.1 Objectives

The objectives of the overall facility preoperational and operational testing programs are to ensure that items relied on for safety:

A. Have been adequately designed and constructed

B. Meet contractual, regulatory, and licensing requirements

C. Do not adversely affect worker or the public health and safety

D. Can be operated in a dependable manner so as to perform their intended function.

Additionally, the preoperational and operational testing programs ensure that operating and emergency procedures are correct and that personnel have acquired the correct level of technical expertise.

Periodic testing at the facility consists of that testing conducted on a periodic basis to monitor various facility parameters and to verify the continuing integrity and capability of IROFS.

Special testing at the facility consists of that testing which does not fall under any other testing program. This testing is of a non-recurring nature and is intended to enhance or supplement existing operational testing rather than replace or supersede other testing or testing programs.

11.2.4.2 Procedure Content

Test Procedures are sufficiently detailed that qualified personnel can perform the required functions without direct supervision. The content of test procedures is uniform to the extent practicable and consists of the following:

A. Title

Each procedure contains a title descriptive of the activities to which it applies.

B. Purpose

The purpose for which the procedure is intended is stated. This statement of applicability is as clear and concise as practicable.

C. References

References are made to specific material used in the preparation and performance of a procedure. This includes applicable drawings, instruction manuals, specifications, and sections of the facility's operating license. These references are listed in a manner as to allow ready location of the material.

D. Time Required

As applicable, estimates of the manpower and time requirements for performance of the specified testing activity are indicated.

E. Prerequisites

Each procedure specifies those items that are required to be completed prior to the performance of the specified testing (e.g., a previous test or special operating conditions). This listing also includes any tests that are to be performed concurrently with the specified testing. Provisions are made to document verification of the completion of the specified prerequisite tests.

F. Test Equipment

Each procedure contains a listing of special test equipment required in performing the specified testing. Procedures contain information and/or references for the items listed such as instruction manuals or procedures.

G. Limits and Precautions

Limits on parameters being controlled and corrective measures necessary to return a parameter to its normal control band are specified. Procedures specifically incorporate limits and corrective measures for all operations affecting criticality safety.

Precautions are specified which alert the individual performing the task, of those situations for which important measures need to be taken early, or where extreme care must be used to protect personnel and equipment or to avoid an abnormal or an emergency situation.

H. Required Plant Unit Status

The procedure specifies the plant unit status necessary to perform the specified testing. Provisions are made to document compliance with the status specified.

I. Prerequisite System Conditions

The procedure specifies the prerequisite system conditions necessary to perform the specified testing. Provisions are made to document compliance with the conditions specified.

J. Test Method

Each procedure contains a brief descriptive section that summarizes the method to be used for performing the specified testing.

K. Data Required

Each procedure specifies any data that must be compiled in the performance of the specified testing in order to verify satisfactory completion of the specified testing. This includes a description of any calculations necessary to reduce raw data to a workable form.

L. Acceptance Criteria

Each procedure states the criteria for evaluating the acceptability of the results of the specified testing. Test results are reduced to a meaningful and readily understandable form in order to facilitate evaluation of their acceptability. Adequate provisions are made to allow documentation of the acceptability, or unacceptability, of test results.

M. Procedure

Procedures contain step-by-step directions in the degree of detail necessary for performing the required testing. References to documents other than the subject procedure are included, as applicable. However, references are identified within these step-by-step directions when the sequence of steps requires that other tasks (not specified by the subject procedure) be performed prior to or concurrent with a procedure step. Where witnessing of a test is required, adequate provisions are made in the test procedure to allow for the required witnessing and to document the witnessing. Cautionary notes, applicable to specific steps, are included and are distinctly identified.

N. Enclosures

Data sheets, checklists and diagrams are attached to the procedure. In particular, checklists utilized to avoid or simplify lengthy or complex procedures are attached as enclosures.

11.2.4.3 Preoperational Testing Program

Preoperation functional tests are completed prior to UF_6 introduction. Other preoperational tests, not required prior to UF_6 introduction and not related to IROFS, such as office building ventilation tests, may be completed following UF_6 introduction. Tests (or portions of tests), which are not required to be completed before UF_6 introduction are identified in the test plan.

The Preoperational testing program comprises three parts:

- Constructor turnover
- Preoperational functional testing
- Initial start up testing.

Constructor Turnover

The constructor is responsible for completion of all as-built drawing verification, purging, cleaning, vacuum testing, system turnover and initial calibration of instrumentation in accordance with design and installation specifications provided by the architect engineers and vendors. As systems or portions of systems are turned over to LES, preoperational testing shall begin. The Technical Services Manager is responsible for coordination of the preoperational and startup test program.

The preoperational test plan including test summaries for all systems is available to the NRC at least 90 days prior to the start of testing. Subsequent changes to the preoperational test plan are also made available to the NRC. Preoperational testing as a minimum includes all system or component tests required by the pertinent design code which were not performed by the constructor prior to turnover. In addition, preoperational tests include all testing necessary to demonstrate that the IROFS are capable of performing their intended function.

Functional Testing

Preoperational functional testing at the facility consists of that testing conducted to initially determine various facility parameters and to initially verify the capability of SSC to meet performance requirements. The tests conducted are primarily associated with IROFS (QA Level 1) and certain QA Level 2 structures, systems and components, but may also include a number of other tests of a technical or financial interest to LES.

Preoperational functional tests are performed following constructor turnover. The major objective of preoperational functional testing is to verify that IROFS essential to the safe operation of the plant are capable of performing their intended function.

For structures, systems and components that are not QA Level 1, acceptance criteria are established to ensure worker-safety Occupational Safety and Health Administration (OSHA), reliable and efficient operation of the system and to demonstrate the performance of intended functions.

Initial startup testing at the facility consists of that testing which includes initial UF_6 introduction and all subsequent testing through the completion of Enrichment Setting Verification for each cascade. "Enrichment Setting Verification" is the verification of a selected enrichment weight percent by measurement of a physical sample collected during the "Enrichment Setting Verification" test run.

Initial startup testing is performed beginning with the introduction of UF_6 and ending with the start of commercial operation. The purpose of initial startup testing is to ensure safe and orderly UF_6 feeding and to verify parameters assumed in the ISA. Examples of initial startup tests include passivation and the filling phase.

Records of the preoperational and startup tests required prior to operation are maintained. These records include testing schedules and the testing results for all IROFS.

Initial Startup Testing

All aspects of initial startup testing are conducted under appropriate test procedures. See Section 11.4, Procedures Development and Implementation, for a detailed description of facility procedures. The use of properly reviewed and approved test procedures is required for all preoperational and startup tests. The results of each preoperational test are reviewed and

approved by the responsible department manager or designee before they are used as the basis of continuing the test program. The results of startup testing are reviewed and approved by the Technical Services Manager. In addition, the results of each individual startup test will receive the same review as that described for preoperation functional tests. All modifications to IROFS that are found necessary are subjected to an evaluation per 10 CFR 70.72 (CFR, 2003e) prior to making the change.

The impact of modifications on future and completed testing is evaluated during the 10 CFR 70.72 (CFR, 2003e) evaluation process and retesting is conducted as required.

Copies of approved test procedures are made available to NRC personnel approximately 60 days prior to their intended use, and not less than 60 days prior to the scheduled introduction of UF_6 for startup tests.

The overall preoperational functional testing program is reviewed, prior to initial UF_6 introduction, by the Plant Manager and all department Managers to ensure that all prerequisite testing is complete.

The facility operating, emergency and surveillance procedures are use-tested throughout the testing program phases and are also used in the development of preoperation functional testing and initial startup testing procedures to the extent practicable. The trial use of operating procedures serves to familiarize operating personnel with systems and plant operation during the testing phases and also serves to ensure the adequacy of the procedures under actual or simulated operating conditions before plant operation begins.

Procedures which cannot be use-tested during the testing program phase are revised based on initial use-testing, operating experience and comparison with the as-built systems. This ensures that these procedures are as accurate and comprehensive as practicable.

11.2.4.4 Operational Testing Program

The operational testing program consists of periodic testing and special testing. Periodic testing is conducted at the facility to monitor various facility parameters and to verify the continuing integrity and capability of facility IROFS. Special testing which may be conducted at the facility is testing which does not fall under any other testing program and is of a non-recurring nature.

The Maintenance Manager has overall responsibility for the development and conduct of the operational testing program and in conjunction with the Health, Safety and Environment (HS&E) Manager ensures that all testing commitments and applicable regulatory requirements are met.

The HS&E Manager shall ensure that new surveillance requirements or testing commitments are identified to the Maintenance Manager. The Maintenance Manager shall make responsibility assignments for new testing requirements.

Surveillance commitments, procedures identified to satisfy these commitments and surveillance procedure responsibility assignments for the facility are identified in a computer database. The database is also used to ensure surveillance testing is completed in the required time interval for all departments.

Test Coordinators are also used for operational testing. The Test Coordinator has the responsibility to be thoroughly familiar with the procedure to be performed. The Test Coordinator should have an adequate period of time in which to review the procedure and the

associated system before the start of the test. It is the responsibility of the appropriate section or department head to designate and ensure that each Test Coordinator meets the appropriate requirements. Operational testing is usually performed by each shift. The Test Coordinator, as part of the shift personnel, also performs regular shift duties in performance of the tests.

The Test Coordinator has the following responsibilities regarding the conduct of testing:

A. Verification of all system and plant unit prerequisites

B. Observance of all limits and precautions during the conduct of the test

C. Compliance with the requirements of the facility license and any other facility directives regarding procedure changes and documentation

D. Identifying and taking corrective actions necessary to resolve system deficiencies or discrepancies observed during the conduct of the test

E. Verification of proper data acquisition, evaluation or results, and compliance with stated acceptance criteria

F. Ensuring that adequate personnel safety precautions are observed during the conduct of the test

G. Coordinating and observing additional manpower and support required from other departments or organizations.

Periodic and special testing procedures are sufficiently detailed that qualified personnel can perform the required functions without direct supervision. The administration requirements for periodic and special testing procedures are the same as ones used for preoperational functional test and initial startup test procedures as identified in Section 11.2.4.3, Preoperational Testing Program. Spaces for initials and dates are required for the following sections:

A. Prerequisite Tests

B. Required Facility (or Plant Unit) Status

C. Prerequisite System Conditions

D. Procedure

E. Enclosures (where calculations are made).

Whenever possible generic procedures and enclosures for recording data for periodic and special tests are used. Also whenever possible, the enclosure is designed as a self-sufficient document that can be filed as evidence that the subject test was performed. Enclosures used as self-sufficient documents should contain sign-off blanks (Initials/Date) to verify that prerequisite tests, required facility status and prerequisite facility or plant unit status and prerequisite system conditions are met before conduct of the test.

11.2.4.4.1 Periodic Testing

The periodic testing program at the facility consists of testing conducted on a periodic basis to verify the continuing capability of IROFS to meet performance requirements.

The facility periodic test program verifies that the facility:

A. Complies with all regulatory and licensing requirements

B. Does not endanger health and minimizes danger to life or property

C. Is capable of operation in a dependable manner so as to perform its intended function.

The facility periodic testing program begins during the preoperational testing stage and continues throughout the facility's life.

A periodic testing schedule is established to ensure that all required testing is performed and properly evaluated on a timely basis. The schedule is revised periodically, as necessary, to reflect changes in the periodic testing requirements and experience gained during plant operation. Testing is scheduled such that the safety of the plant is never dependent on the performance of an IROFS that has not been tested within its specified testing interval.

Periodic test scheduling is handled through the Maintenance department. The Maintenance department maintains the periodic test status index on the computer database. The purpose of this index is to assist groups in assuring that all surveillances are being completed within the required test interval.

The database includes all periodic testing, calibration or inspection required by regulatory requirements or licensing commitments, and provides the following information for each surveillance:

- Test #
- Title
- Equipment #
- Work Request # (if applicable)
- Test Frequency
- Plant Cascade #
- Last date test was performed
- Next date test is due.

In the event that a test cannot be performed within its required interval due to system or plant unit conditions, the responsible department notifies in writing, on the applicable form, the HS&E Manager, Operations Manager, and the Maintenance Manager, as appropriate. The originating department retains a copy as a record of the transmittal. The responsible department lists the earliest possible date the test could be performed and the latest date along with the required system or unit-mode condition. However, the responsible department will ensure that the test is performed as soon as practical once required conditions are met, regardless of the estimated date given earlier.

Periodic testing and surveillance associated with QA Level 1 and 2 structures, systems and components are performed in accordance with written procedures.

11.2.4.4.2 Special Testing

Special testing is testing conducted at the facility that is not a facility preoperational test, periodic test, post-modification test, or post-maintenance test. Special testing is of a non-recurring nature and is conducted to determine facility parameters and/or to verify the capability of IROFS to meet performance requirements. Purposes of special testing include, but are not necessarily limited to, the following:

A. Acquisition of particular data for special analysis

B. Determination of information relating to facility incidents

C. Verification that required corrective actions reasonably produce expected results and do not adversely affect the safety of operations

D. Confirmation that facility modifications reasonably produce expected results and do not adversely affect systems, equipment and/or personnel by causing them to function outside established design conditions; applicable to testing performed outside of a post-modification test.

The determination that a certain plant activity is a Special Test is intended to exclude those plant activities which are routine surveillances, normal operational evolutions, and activities for which there is previous experience in the conduct and performance of the activity. At the discretion of the Plant Manager, any test may be conducted as a special test. In making this determination, facility management includes the following evaluations of characteristics of the activity:

A. Does the activity involve an unusual operational configuration for which there is no previous experience?

B. Does the activity have the propensity, if improperly conducted, to significantly affect primary plant parameters?

C. Does the activity involve seldom-performed evolutions, meeting one of the above criteria, in which the time elapsed since the previous conduct of the activity renders prior experience not useful?

11.3 TRAINING AND QUALIFICATIONS

This section describes the training program for the operations phase of the facility, including preoperational functional testing and initial startup testing. The training program requirements apply to those plant personnel who perform activities relied on for safety.

The QA Program provides training and qualification requirements, during the design, construction, and operations phases, for QA training of personnel performing QA levels 1 and 2 work activities; for nondestructive examination, inspection, and test personnel; and for QA auditors.

The principle objective of the LES training program system is to ensure job proficiency of all facility personnel involved in work through effective training and qualification. The training program system is designed to accommodate future growth and meet commitments to comply with applicable established regulations and standards.

Qualification is indicated by successful completion of prescribed training, demonstration of the ability to perform assigned tasks and where required by regulation, maintaining a current and valid license issued by the agency establishing the requirements. Training is designed, developed and implemented according to a systematic approach. Employees are provided with formal training to establish the knowledge foundation and on-the-job training to develop work performance skills. Continuing training is provided, as required, to maintain proficiency in these knowledge and skill components, and to provide further employee development.

The training program described in this section is consistent with that previously submitted for NRC review in Section 11.3 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC Staff reviewed the previous submittal and found it to be acceptable. The NRC staff's review and conclusions associated with Training are documented in Section 10.4 of NUREG-1491 (NRC, 1994).

11.3.1 Organization and Management of the Training Function

Line managers are responsible for the content and effective conduct of training for their personnel. Training responsibilities for line managers are included in position descriptions, and line managers are given the authority to implement training for their personnel. The training organization provides support to line managers by facilitating the planning, directing, analyzing, developing, conducting, evaluating, and controlling of a systematic performance-based training process. Performance-based training is used as the primary management tool for analyzing, designing, developing, conducting, and evaluating training.

Facility administrative procedures establish the requirements for indoctrination and training of personnel performing activities relied on for safety and to ensure that the training program is conducted in a reliable and consistent manner throughout all training areas. Exceptions from training requirements may be granted when justified and documented in accordance with procedures and approved by appropriate management.

Lesson plans are used for classroom and on-the-job training to provide consistent subject matter. When design changes or facility modifications are implemented, updates of applicable lesson plans are included in the change control process of the configuration management program.

Training records are maintained to support management information needs associated with personnel training, job performance, and qualifications.

The training programs at the facility are the responsibility of the Human Resources Manager. Records are 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 facility. The employee training file shall also contain records of special company sponsored training conducted by others. The training records for each individual are maintained so that they are accurate and retrievable. Training records are retained in accordance with the records management procedures.

11.3.2 Analysis and Identification of Functional Areas Requiring Training

A needs/job analysis is performed and tasks are identified to ensure that appropriate training is provided to personnel working on tasks related to IROFS. Additionally, Job Hazard Analysis (JHA), sometimes referred to as Job Safety Analysis (JSA) (i.e., a step-by-step process used to evaluate job hazards), will be used as part of on-the-job training for providing employees the skills necessary to perform their jobs safely at the NEF.

The training organization consults with relevant technical and management personnel as necessary to develop a list of tasks for which personnel training for specific jobs is appropriate. The list of tasks selected for training is reviewed and compared to the training materials as part of the systematic evaluation of training effectiveness. The task list is also updated as necessitated by changes in procedures, processes, plant systems, equipment, or job scope.

11.3.3 Position Training Requirements

Minimum training requirements are developed for those positions whose activities are relied on for safety. Initial identification of job-specific training requirements is based on experience. Entry-level criteria (e.g., education, technical background, and/or experience) for these positions are contained in position descriptions.

The training program is designed to prepare initial and replacement personnel for safe, reliable and efficient operation of the facility. Appropriate training for personnel of various abilities and experience backgrounds is provided. The level at which an employee initially enters the training program is 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 that consists of the following:

- General Employee Training
- Technical Training
- Employee Development/Management-Supervisory Training.

Training is made available to facility personnel to initially develop and maintain minimum qualifications outlined in Chapter 2, Organization and Administration. The objective of the training shall be to ensure safe and efficient operation of the facility and compliance with applicable established regulations and requirements. 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, testing or other technical aspect of the facility IROFS. Training courses are kept up-to-date to reflect plant modifications and changes to procedures when applicable.

Continuing or periodic retraining courses shall be established when applicable to ensure that personnel remain proficient. Periodic retraining generally is conducted to ensure retention of knowledge and skills important to facility operations. The training may consist of periodic retraining exercises, instruction, and review of subjects as appropriate to maintain proficiency of all personnel assigned to the facility. Section 7, Maintenance of Radiological Contingency Preparedness Capability, of the Emergency Plan provides additional information on personnel training for emergency response tasks.

11.3.3.1 General Employee Training

General Employee Training encompasses those Quality Assurance, radiation protection, safety, emergency and administrative procedures established by facility management and applicable regulations. The safety training for the NEF complies with the applicable sections of Occupational Safety and Health Administration (OSHA) regulations such as 29 CFR 1910 (Occupational Safety and Health Standards), 1910.1200 (Hazard Communication), and with NRC regulations such as 10 CFR 20 (Standards for Protection Against Radiation) and 10 CFR 19 (Notices, Instructions and Reports to Workers: Inspection and Investigations). Continuing training is conducted in these areas as necessary to maintain employee proficiency. All persons under the supervision of facility management (including contractors) must participate in General Employee Training; however, certain facility support personnel, depending on their normal work assignment, may not participate in all topics of this training. Temporary maintenance and service personnel receive General Employee Training to the extent necessary to assure safe execution of their duties. Certain portions of General Employee Training may be included in a New Employee Orientation Program.

General Employee Training topics are listed below:

- General administrative controls and procedure use
- Quality Assurance policies and procedures
- Facility systems and equipment
- Nuclear safety (See Section 11.3.3.1.1 - 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)
- Chemical Safety
- Fire Protection and Fire Brigade (see Section 11.3.3.1.2)
- New Employee Orientation.

11.3.3.1.1 Nuclear Safety Training

Training programs are established for the various types of job functions (e.g., production operator, radiation protection technician, contractor personnel) commensurate with criticality safety and/or radiation safety responsibilities associated with each such position. Visitors to the Controlled Access Area are trained in the formal training program or are escorted by trained personnel while in the Controlled Access Area.

This training is highlighted to stress the high level of importance placed on the radiological, criticality and chemical safety of plant personnel and the public. This training is structured as follows:

A. Personnel access procedures ensure the completion of formal nuclear safety training prior to permitting unescorted access into the Controlled Access Area.

B. Training sessions covering criticality safety, radiation protection and emergency procedures are conducted on a regular basis to accommodate new employees or those requiring retraining. Topics covered in the training program include:

- Notices, reports and instructions to workers
- Practices designed to keep radiation exposures ALARA
- Methods of controlling radiation exposures
- Contamination control methods (including decontamination)
- 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
- Radiation protection practices
- Protective clothing
- Respiratory protection
- Personnel surveys.

Criticality safety training shall be in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996) and ANSI/ANS-8.20-1991 (ANSI, 1991).

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 and assessments of operations and maintenance personnel responsible for following the requirements related to the topics listed above.

Newly hired or transferred employees reporting for work prior to the next regularly scheduled training session must complete nuclear safety training prior to unescorted access into the Controlled Access Area.

Since contractor employees perform diverse tasks in the Controlled Access Area, formal training for these employees is 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 Controlled Access Area.

These training programs are conducted by instructors assigned by the HS&E Manager as having the necessary knowledge to address criticality safety and radiation protection. Records of the training programs are maintained as described in Section 11.7, “Records Management.”

C. Individuals requiring unescorted access to the Controlled Access Area receive annual retraining. Retraining for individuals is scheduled and reported by means of a computerized tracking system.

D. Contents of the formal nuclear safety training programs are reviewed and updated periodically by the HS&E Manager, or designee, to ensure that the programs are current and adequate. In addition, at least annually, the contents of the radiation protection sections of the nuclear safety training program are reviewed and updated, as required, by the HS&E Manager or his designee.

E. Operational personnel are 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 are reviewed with operational personnel by their immediate supervisor or delegate.

F. Radiation safety topics are also 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.

11.3.3.1.2 Fire Brigade Training

The primary purpose of the Fire Brigade Training Program is to develop a group of facility employees skilled in fire prevention, fire fighting techniques, first aid procedures, and emergency response. They are trained and equipped to function as a team for the fighting of fires. The intent of the facility fire brigade is 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 provides 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.

11.3.3.2 Technical Training

Technical training is 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 is used to develop manipulative skills necessary to perform assigned work in a competent manner. Technical training consists of four segments:

- Initial Training
- On-the-Job Training and Qualifications
- Continuing Training
- Special Training.

11.3.3.2.1 Initial Training

Initial job training is 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 is not limited to, live lectures, taped and filmed lectures, self-guided study, demonstrations, laboratories and workshops 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 is 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 are developed for operations, maintenance and technical services classifications. Training for each program is grouped into logical blocks or modules and presented in such a manner that specific behavioral objectives are accomplished. Trainee progress is evaluated using written examinations, oral or practical tests. Depending upon the regulatory requirements or individual's needs and plant operating conditions, allowances are made to suit specific situations. Brief descriptions of modules that may be contained in the initial training programs are as follows:

<u>Operations Initial Training</u>

A. General Systems

This training module provides the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components are taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment facility.

B. Specific Systems

This training module provides basic instruction in system and component identification and basic system operating characteristics. It provides a general overview of enrichment plant equipment and acquaints the trainees with enrichment plant

terminology and nomenclature and provides instruction describing basic system operations.

C. Nuclear Preparatory

This training module develops the necessary concepts in basic nuclear physics, plant chemistry, basic thermodynamics, radiation protection, and enrichment theory. Experience in enrichment control and radiation protection is also provided. It is normally presented to operations personnel following the Systems Specific training module.

D. Plant Familiarization

The Plant Familiarization module provides for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the facility.

Mechanical Maintenance Initial Training

A. General Systems

This training module provides the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components are taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment facility.

B. Fundamental Shop Skills

This training module provides instruction in fundamentals of mechanical maintenance performance. It combines academic instruction with hands-on training to familiarize trainees with design operational and physical characteristics of enrichment facility components, and basic skills and procedures used to perform mechanical repairs and/or equipment replacement. Task training lists are integrated into this module to assure that each trainee attains a minimum level of performance. Tasks are assigned and trainees use work procedures to guide them through a task. Both radiological and industrial safety is stressed in all phases of this training module.

C. Plant Familiarization

The Plant Familiarization module provides for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the facility.

Instrumentation and Electrical and Maintenance Initial Training

A. General Systems

This training module provides the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components are taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment facility.

B. Basic Instrument and Electrical

This training module provides 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 facilities.

The module also provides 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 familiarizes the trainee with plant test procedures, test equipment, and testing as well as plant records, reports, and data collection. It provides a basic understanding of thermodynamics used in testing plant heat transfer.

D. Plant Familiarization

The Plant Familiarization module provides for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the plant.

Health Physics and Chemistry Initial Training

A. General Systems

This training module provides the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components are taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment facility.

B. Fundamental Health Physics

The Fundamental Health Physics Module presents to the trainees a more comprehensive and theoretical understanding of the nuclear processes with which they are involved. In addition, the techniques for applying theory are presented in this module. Use is made of various non-automated counting and spectrographic equipment and portable survey instruments. Administrative material is also presented in a more detailed manner.

C. Fundamental Chemistry

The Fundamental Chemistry module provides familiarization with chemistry theory, techniques, and procedures. The overall goal of this module is familiarization necessary for chemistry technicians to be able to work safely and competently in the enrichment facility.

D. Plant Familiarization

The Plant Familiarization module provides for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the plant.

Engineer/Professional Initial Training

This training is part of the technical staff and managers training program.

A. Facility Orientation

This training module provides an orientation to each section within the NEF. An on-the-job task list provides the trainee with training objectives that must be accomplished while working in the section.

B. Basic Engineer/Professional Training

The Basic Engineer/Professional Training provides a 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 provides specific theoretical information related to enrichment plant operations. Topics (e.g., Thermal Science, Nuclear Physics) address applications in an enrichment facility.

D. Engineer/Professional Systems Training

The Engineer/Professional Systems Training provides an overview of plant systems, components and procedures necessary to operate an enrichment plant safely and efficiently.

11.3.3.2.2 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 is conducted in the work environment. Applicable tasks and related procedures make up the OJT/qualifications program for each technical area which is designed to supplement and complement training received through formal classroom, laboratory, and/or simulator training. The objective of the program is to assure the trainee's ability to perform job tasks as described in the task descriptions and the Training and Qualification Guides.

11.3.3.2.3 Continuing Training

Continuing training is any training not provided as initial qualification and basic training which maintains and improves job-related knowledge and skills such as the following:

- Facility systems and component changes
- OJT/Qualifications program retraining
- Policy and procedure changes
- Operating experience program documents review to include Industry and in-house operating experiences
- Continuing training required by regulation (e.g., emergency plan training)
- General employee, special, administrative, vendor, and/or advanced training topics supporting tasks that are elective in nature

- Training identified to resolve deficiencies (task-based) or to reinforce seldom used knowledge skills
- Refresher training on initial training topics
- Structured pre-job instruction, mock-up training, and walk throughs
- Quality awareness.

Continuing Training and Retraining may overlap to some degree in definition; however, Retraining refers to specific training designed for proficiency maintenance.

Continuing Training consists of formal and informal components performed on a frequency needed to maintain proficiency on the job. Each Section's Continuing Training Program is developed from a systematic approach, using information from job performance and safe operation as a basis for determining the content of continuing training. Continuing training may be offered, as needed, on any of the topics listed above.

Once the objectives for Continuing Training have been established, the methods for conducting the training may vary. The method selected must provide clear evidence of objective accomplishment and consistency in delivery.

11.3.3.2.4 Special Training

Special training involves those subjects of a unique nature required for a particular area of work. Special training is usually given to selected personnel based on specific needs not directly related to disciplinary lines.

11.3.4 Basis and Objectives for Training

Learning objectives identify the training content, as established by needs/job analyses and position-specific requirements. The task list from the needs/job analysis is used to develop action statements that describe the desired post-training performance. Objectives include the knowledge, skills, and abilities the trainee should demonstrate; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

11.3.5 Organization of Instruction, Using Lesson Plans and Other Training Guides

Lesson plans are developed from the learning objectives that are based on job performance requirements. Lesson plans and other training guides are developed under the guidance of the training function. Lesson plans are reviewed by the training function and, generally, by the organization cognizant of the subject matter. Lesson plans are approved prior to issue or use. Lesson plans are used for classroom training and on-the-job training as required and include Standards for evaluating acceptable trainee performance.

11.3.6 Evaluation of Trainee Learning

Trainee understanding and command of learning objectives is evaluated through observation/demonstration or oral or written tests as appropriate. Such evaluations measure the trainee's skills and knowledge of job performance requirements.

Evaluations are performed by individuals qualified in the training subject matter.

11.3.7 Conduct of On-the-Job Training

On-the-Job Training is an element of the technical training program (see Section 11.3.3.2.2, On-the-Job Training and Qualifications). On-the-job training is used in combination with classroom training for activities that are IROFS. Designated personnel who are competent in the program standards and methods of conducting the training conduct on-the-job training using current performance-based training materials. Completion of on-the-job training is demonstrated by actual task performance or performance of a simulation of the task with the trainee explaining task actions using the conditions encountered during the performance of the task, including references, tools, and equipment reflecting the actual task to the extent practical.

11.3.8 Evaluation of Training Effectiveness

Periodically the training program is systematically evaluated to measure the program's effectiveness in producing competent employees. The trainees provide feedback after completion of classroom training sessions to provide data for this evaluation for program improvements. These evaluations identify program strengths and weaknesses, determine whether the program content matches current job needs, and determine if corrective actions are needed to improve the program's effectiveness. The training function is responsible for leading the training program evaluations and for implementing any corrective actions. Program evaluations may consist of an overall periodic evaluation or a series of topical evaluations over a given period.

Evaluation objectives that are applicable to the training program or topical area being reviewed are developed and may address the following elements of training:

- Management and administration of training and qualification programs
- Development and qualification of the training staff
- Position training requirements
- Determination of training program content, including its facility change control interface with the configuration management system
- Design and development of training programs, including lesson plans
- Conduct of training
- Trainee examinations and evaluations

- Training program assessments and evaluations.

Evaluation results are documented, with program strengths and weaknesses being highlighted. Identified weaknesses are reviewed, improvements are recommended, and changes are made to procedures, practices, or training materials as necessary.

Periodically, training and qualifications activities are monitored by designated facility and/or contracted training personnel. The Quality Assurance Department audits the facility training and qualification system. In addition, trainees and vendors may provide input concerning training program effectiveness. Methods utilized to obtain this information include, among other things surveys, questionnaires, performance appraisals, staff evaluation, and overall training program effectiveness evaluation instruments. Frequently conducted classes are not evaluated each time. However, they are routinely evaluated at a frequency sufficient to determine program effectiveness. Evaluation information may be collected through:

- Verification of program objectives as related to job duties for which intended
- Periodic working group program evaluations
- Testing to determine trainee accomplishment of objectives
- Trainee 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 is transmitted to the appropriate Line Manager.

11.3.9 Personnel Qualification

The qualification requirements for key management positions are described in Chapter 2, Organization and Administration. Training and qualification requirements associated with QA personnel are provided in Appendix A to this chapter. In addition, qualification and training requirements for process operator candidates shall be established and implemented in plant procedures.

11.3.10 Periodic Personnel Evaluations

Personnel performing activities relied on for safety are evaluated at least biennially to determine whether they are capable of continuing their activities that are relied on for safety. The evaluation may be by written test, oral test, or on-the-job performance evaluation. The results of the evaluation are documented. When the results of the evaluation dictate, retraining or other appropriate action is provided. Retraining is also required due to plant modifications, procedure changes, and QA program changes that result in new or revised information.

11.4 PROCEDURES DEVELOPMENT AND IMPLEMENTATION

The management measure described in this section is consistent with that previously submitted for NRC review in Section 11.4 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC staff reviewed the previous submittal and found it to be acceptable. The NRC staff's review and conclusions associated with procedures are documented in Section 10.5 of NUREG-1491 (NRC, 1994).

The requirements for independent verification are consistent with the applicable guidance provided in ANSI/ANS-3.2-1994, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."

All activities involving licensed materials or IROFS are conducted in accordance with approved procedures. Before initial enrichment activities occur at the facility, procedures are made available to the NRC for their inspection. As noted throughout this document, procedures are used to control activities in order to ensure the activities are carried out in a safe manner and in accordance with regulatory requirements.

Generally, four types of plant procedures are used to control activities: operating procedures, administrative procedures, maintenance procedures, and emergency procedures.

Operating procedures, developed for workstation and Control Room operators, are used to directly control process operations. Operating procedures include:

- Purpose of the activity
- Regulations, polices, and guidelines governing the procedure
- Type of procedure
- Steps for each operating process phase:
    - Initial startup
    - Normal operations
    - Temporary operations
    - Emergency shutdown
    - Emergency operations
    - Normal shutdown
    - Startup following an emergency or extended downtime.
- Hazards and safety considerations
- Operating limits
- Precautions necessary to prevent exposure to hazardous chemicals (resulting from operations with Special Nuclear Material (SNM)) or to licensed SNM.

- Measures to be taken if contact or exposure occurs
- IROFS associated with the process and their functions
- The timeframe for which the procedure is valid.

Applicable safety limits and IROFS are clearly identified in the procedures. LES will incorporate methodology for identifying, developing, approving, implementing, and controlling operating procedures. Identifying needed procedures will include consideration of ISA results. The method will ensure that, as a minimum:

- Operating limits and IROFS are specified in the procedure
- Procedures include required actions for off-normal conditions of operation, as well as normal operations
- If needed safety checkpoints are identified at appropriate steps in the procedure
- Procedures are validated through field tests
- Procedures are approved by management personnel responsible and accountable for the operation
- A mechanism is specified for revising and reissuing procedures in a controlled manner
- The QA elements and CM Program at the facility provide reasonable assurance that current procedures are available and used at all work locations
- The facility training program trains the required persons in the use of the latest procedures available.

Administrative procedures are used to perform activities that support the process operations, including management measures such as the following:

- Configuration management
- Nuclear criticality, radiation, chemical, and fire safety
- Quality Assurance
- Design control
- Plant personnel training and qualification
- Audits and assessments
- Incident investigations
- Record keeping and document control

- Reporting
- Procurement.

Administrative procedures are also used for:

- Implementing the Fundamental Nuclear Material Control (FNMC) Plan
- Implementing the Emergency Plan
- Implementing the Physical Security Plan
- Implementing the Standard Practice Procedures Plan for the Protection of Classified Matter.

Maintenance procedures address:

- Preventive and corrective maintenance of IROFS
- Surveillance (includes calibration, inspection, and other surveillance testing)
- Functional testing of IROFS
- Requirements for pre-maintenance activity involving reviews of the work to be performed and reviews of procedures.

Emergency procedures address the preplanned actions of operators and other plant personnel in the event of an emergency.

Procedures will be established and implemented for nuclear criticality safety in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996). The NCS procedures will be written such that no single, inadvertent departure from a procedure could cause an inadvertent criticality. Nuclear criticality safety postings at the NEF are established that identify administrative controls applicable and appropriate to the activity or area in question. Nuclear criticality safety procedures and postings are controlled by procedure to ensure that they are maintained current.

Periodic reviews will be performed on procedures to assure their continued accuracy and usefulness. Specifically, reviews of operating procedures will be conducted at a minimum of every five years and reviews of radiation protection procedures and emergency procedures will be conducted at a minimum of every year. In addition, applicable procedures will be reviewed after unusual incidents, such as an accident, unexpected transient, significant operator error, or equipment malfunction, or after any modification to a system, and procedures will be revised as needed.

11.4.1 Preparation of Procedures

Each procedure is assigned to a member of the facility staff or contractor for development. Initial procedure drafts are reviewed by other appropriate members of the facility staff, by personnel from the supplier of centrifuges (Urenco), and other vendors, as appropriate for inclusion and correctness of technical information, including formulas, set points, and acceptance criteria and includes either a walkdown of the procedure in the field or a tabletop walkthrough. Procedures that are written for the operation of IROFS shall be subjected to an independent review. The designated approver shall determine whether or not any additional,

cross-disciplinary review is required. The Plant Manager or designee shall approve all procedures. If the procedure involves QA directly, the QA Manager must approve the procedure.

11.4.2 Administrative Procedures

Facility administrative procedures are written by each department as necessary to control activities that support process operations, including management measures. Listed below are several areas for which administrative procedures are written, including principle features:

A. Operator's authority and responsibility: The operator is 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 Manager. The operators are also assigned the responsibility for knowing the limits and set points associated with safety-related equipment and systems as specified in designated operating procedures.

B. Activities affecting facility operation or operating indications: All facility maintenance personnel performing support functions (e.g., maintenance, testing) which may affect unit operation or Control Room indications are required to notify the Control Room Operator and/or Shift Manager, as appropriate, prior to initiating such action.

C. Manipulation of facility control: No one is permitted to manipulate the facility controls who is not an operator, except for operator trainees under the direction of a qualified operator.

D. Relief of Duties: This procedure provides a detailed checklist of applicable items for shift turnover.

E. Equipment control: Equipment control is maintained and documented through the use of tags, labels, stamps, status logs or other suitable means.

F. Master surveillance testing schedule: A master surveillance testing schedule is documented to ensure that required testing is performed and evaluated on a timely basis. Surveillance testing is 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 identifies surveillance and testing requirements, applicable procedures, and required test frequency. Assignment of responsibility for these requirements is also indicated.

G. A Control Room Operations Logbook is maintained. This logbook contains significant events during each shift such as enrichment changes, alarms received, or abnormal operational conditions.

H. Fire Protection Procedures: Fire protection procedures are written to address such topics as training of the fire brigade, reporting of fires, and control of fire stops. The facility's Industrial Safety department has 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.

The administrative control of maintenance is maintained as follows:

A. In order to assure safe, reliable, and efficient operation, a comprehensive maintenance program for the facility's IROFS is established.

B. Personnel performing maintenance activities are qualified in accordance with applicable codes and standards and procedures.

C. Maintenance is performed in accordance with written procedures that conform to applicable codes, standards, specifications, and other appropriate criteria.

D. Maintenance is scheduled so as not to jeopardize facility operation or the safety of facility personnel.

E. Maintenance histories are maintained on facility IROFS.

The administrative control of facility modifications is discussed in Section 2.3.1, Configuration Management.

11.4.3 Procedures

All activities involving licensed materials or IROFS are conducted in accordance with approved procedures. These procedures are intended to provide a pre-planned method of conducting operations of systems in order to eliminate errors due to on-the-spot analysis and judgments.

All procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. However, written procedures cannot address all contingencies and operating conditions. Therefore, they contain a degree of flexibility appropriate to the activities being performed. Procedural guidance exists to identify the manner in which procedures are to be implemented. For example, routine procedural actions may not require the procedure to be present during implementation of the actions, while complex jobs, or checking with numerous sequences may require valve alignment checks, approved operator aids, or in-hand procedures that are referenced directly when the job is conducted.

Examples of operating activities are:

- 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 Feed/Take-off Stations and Preparation for Operation
- Removal of UF_6 Cylinder from Feed/Take-off Stations
- Installation of UF_6 Cylinders in Take-off Stations
- UF_6 Gas Sampling in Take-off Lines
- UF_6 Sampling in Product Liquid Sampling Autoclaves

- Emptying of Cold Trap
- Exchange of Chemical Traps in Vent Systems.

Plant specific procedures for abnormal events are written for the facility. These procedures are 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 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 event procedures are based on a conservative course of action to be followed by the operating crew.

Typical abnormal event procedures include:

- Power Failure
- Loss of Heat Tracing
- Damaged UF_6 Cylinder Repairs
- Annunciator alarms (procedures to include alarm set points, probable causes, automatic actions, immediate manual actions, supplementary actions and applicable references).

Temporary changes to procedures are issued for operating activities that are of a nonrecurring nature. Temporary changes to procedures are used when revision of an operating or other permanent procedure is not practical. Temporary changes to procedures shall not involve a change to the ISA and shall not alter the intent of the original procedure. Examples of uses of temporary changes to procedures 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 temporary changes to procedures are approved by two members of the facility management staff, at least one of whom is a shift manager. Temporary changes to procedures are documented, reviewed and approved with the process described in Section 11.4.4, Changes to Procedures, within 14 days of implementation.

Maintenance of facility structures, systems and components is performed in accordance with written procedures, documented instructions, checklists, or drawings appropriate to the circumstances (for example, skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineation in a written procedure) that conform to applicable codes, standards, specifications, and other appropriate criteria.

The facility's maintenance department under the Maintenance Manager has responsibility for preparation and implementation of maintenance procedures. The maintenance, testing and calibration of facility IROFS is 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 IROFS to meet performance requirements is conducted in accordance with approved, written procedures. Periodic test procedures are utilized to perform such testing and are sufficiently detailed that qualified personnel can perform the required functions without direct supervision. Testing performed on IROFS that are not redundant will provide for compensatory measures to be put into place to ensure that the IROFS performs until it is put back into service.

Periodic test procedures are performed by the facility's Technical Services, Operations and Maintenance departments. The Maintenance Manager has overall responsibility for assuring that the periodic testing is in compliance with the requirements.

Chemical and radiochemical activities associated with facility IROFS are performed in accordance with approved, written procedures. The facility's chemistry department has responsibility for preparation and implementation of chemistry procedures.

Radioactive waste management activities associated with the facility's liquid, gaseous, and solid waste systems are performed in accordance with approved written procedures. The facility's operations, chemistry and radiation protection departments have responsibility for preparation and implementation of the radioactive waste management procedures.

Likewise, other departments at the facility develop and implement activities at the facility through the use of procedures.

Procedures will include provisions for operations to stop and place the process in a safe condition if a step of a procedure cannot be performed as written.

11.4.4 Changes to Procedures

Changes to procedures shall be processed as described below.

A. The preparer documents the change as well as the reason for the change.

B. An evaluation shall be performed in accordance with 10 CFR 70.72 (CFR, 2003e) as appropriate. If the evaluation reveals that a change to the license is needed to implement the proposed changes, the change is not implemented until prior approval is received from the NRC.

C. The procedure with proposed changes shall be reviewed by a qualified reviewer.

D. The Plant Manager, a Department Manager, or a designee approved by the Plant Manager shall be responsible for approving procedure changes, and for determining whether a cross-disciplinary review is necessary, and by which department(s). The need for the following cross-disciplinary reviews shall be considered, as a minimum:

   1. For proposed changes having a potential impact on chemical or radiation safety, a review shall be performed for chemical and radiation hazards. Changes shall be approved by the HS&E Manager or designee.

2. For proposed changes having a potential impact on criticality safety, a nuclear criticality safety evaluation shall be performed. Any necessary controlled parameters, limits, IROFS, management measures, or NCS analyses that must be imposed or revised are adequately reflected in appropriate procedures and/or design basis documents. Changes shall be independently reviewed by a criticality safety engineer, and approved by the HS&E Manager or designee.

3. For proposed changes potentially affecting Material Control and Accounting, a material control review shall be performed. Changes shall be approved by the HS&E Manager or designee.

Records of completed cross-functional reviews shall be maintained in accordance with Section 11.7, Records Management, for all changes to procedures involving licensed materials or IROFS.

11.4.5 Distribution of Procedures

Originally issued approved procedures and approved procedure revisions are distributed in a controlled manner by document control.

Document Control shall establish and maintain an index of the distribution of copies of all facility procedures. Revisions are controlled and distributed in accordance with this index. Indexes are reviewed and updated on a periodic basis or as required.

Department Managers or their designees shall be responsible for ensuring all personnel doing work which require the use of the procedures have ready access to controlled copies of the procedures.

11.5 AUDITS AND ASSESSMENTS

LES will have a tiered approach to verifying compliance to procedures and performance to regulatory requirements. Audits are focused on verifying compliance with regulatory and procedural requirements and licensing commitments. Assessments are focused on effectiveness of activities and ensuring that IROFS are reliable and are available to perform their intended safety functions. This approach includes performing Assessments and Audits on critical work activities associated with facility safety, environmental protection and other areas as identified via trends.

Assessments are divided into two categories that will be owned and managed by the line organizations as follows:

- Management Assessments conducted by the line organizations responsible for the work activity
- Independent Assessments conducted by individuals not involved in the area being assessed.

Audits of the QA Level 1 work activities associated with IROFS and any items that affect the function of the IROFS and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied will be the responsibility of the QA Department.

Audits and assessments are performed to assure that facility activities are conducted in accordance with the written procedures and that the processes reviewed are effective. As a minimum, they shall assess activities related to radiation protection, criticality safety control, hazardous chemical safety, industrial safety including fire protection, and environmental protection.

Audits and assessments shall be performed routinely by qualified staff personnel that are not directly responsible for production activities. Deficiencies identified during the audit or assessment requiring corrective action shall be forwarded to the responsible manager of the applicable area or function for action in accordance with the CAP procedure. Future audits and assessments shall include a review to evaluate if corrective actions have been effective.

The Quality Assurance Department shall be responsible for audits. Audits shall be performed in accordance with a written plan that identifies and schedules audits to be performed. Audit team members shall not have direct responsibility for the function and area being audited. Team members shall have technical expertise or 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 in a timely manner to the Plant Manager, the Safety Review Committee (SRC), and the Managers responsible for the activities audited. Any deficiencies noted in the audits shall be responded to promptly by the responsible Managers or designees, entered into the CAP and tracked to completion and re-examined during future audits to ensure corrective action has been completed.

Records of the instructions and procedures, persons conducting the audits or assessments, and identified violations of license conditions and corrective actions taken shall be maintained.

The management measure described in this section and Chapter 2, Organization and Administration, is consistent with that previously submitted for NRC review in the Claiborne

Enrichment Center Safety Analysis Report (LES, 1993). The NRC Staff reviewed the previous submittal and found it to be acceptable. The NRC Staff's review and conclusions associated with audits and assessments are documented in Section 10.7 of NUREG-1491 (NRC, 1994).

11.5.1 Activities to be Audited or Assessed

Audits and assessments are conducted for the areas of:

- Radiation safety
- Nuclear criticality safety
- Chemical safety
- Industrial safety including fire protection
- Environmental protection
- Emergency management
- QA
- Configuration management
- Maintenance
- Training and qualification
- Procedures
- CAP/Incident investigation
- Records management.

Assessments of nuclear criticality safety, performed in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996), will ensure that operations conform to criticality requirements.

11.5.2 Scheduling of Audits and Assessments

A schedule is established that identifies audits and assessments to be performed and the responsible organization assigned to conduct the activity. The frequency of audits and assessments is based upon the status and safety importance of the activities being performed and upon work history. All major activities will be audited or assessed on an annual basis. The audit and assessment schedule is reviewed periodically and revised as necessary to ensure coverage commensurate with current and planned activities.

Nuclear Criticality safety audits are conducted and documented quarterly such that all aspects of the Nuclear Criticality Safety Program will be audited at least every two years. The Operations Group is assessed periodically to ensure that nuclear critical safety procedures are being followed and the process conditions have not been altered to adversely affect nuclear criticality safety. The frequency of these assessments is based on the controls identified in the NCSEs. Assessments are conducted at least semi-annually. In addition, weekly nuclear criticality safety walkthroughs of UF_6 process areas are conducted and documented.

11.5.3 Procedures for Audits and Assessments

Internal and external audits and assessments are conducted using approved procedures that meet the QA Program requirements. These procedures provide requirements for the following audit and assessment activities:

- Scheduling and planning of the audit and assessment
- Certification requirements of audit personnel
- Development of audit plans and audit and assessment checklists as applicable
- Performance of the audit and assessment
- Reporting and tracking of findings to closure
- Closure of the audit and assessment.

The applicable procedures emphasize reporting and correction of findings to prevent recurrence.

Audits and assessments are conducted by:

- Using the approved audit and assessment checklists as applicable
- Interviewing responsible personnel
- Performing plant area walkdowns
- Reviewing controlling plans and procedures
- Observing work in progress
- Reviewing completed QA documentation.

Audit and assessment results are tracked in the Corrective Action Program. The data is periodically analyzed for potential trends and needed program improvements to prevent recurrence and/or for continuous program improvements. The resulting trend is evaluated and reported to applicable management. This report documents the effectiveness of management measures in controlling activities, as well as deficiencies. Deficiencies identified in the trend

report require corrective action in accordance with the applicable CAP procedure. The QA organization also performs follow up reviews on identified deficiencies and verifies completion of corrective actions reported as a result of the trend analysis.

The audit and /or assessment team leader is required to develop the audit and /or assessment report documenting the findings, observations, and recommendations for program improvement. These reports provide management with documented verification of performance against established performance criteria for IROFS. These reports are developed, reviewed, approved, and issued following established formats and protocols detailed in the applicable procedures. Responsible managers are required to review the reports and provide any required responses due to reported findings.

Corrective actions following issuance of the audit and/or assessment report require compliance with the CAP procedure. Audit reports are required to contain an effectiveness evaluation and statement for each of the applicable QA program elements reviewed during the audit. The audit/assessment is closed with the proper documentation as required by the applicable audit and assessment procedure. The QA organization will conduct follow-up audits or assessments to verify that corrective actions were taken in a timely manner. In addition, future assessments will include a review to evaluate if corrective actions have been effective.

11.5.4 Qualifications and Responsibilities for Audits and Assessments

The QA Director or QA Manager initiates audits. The responsible Lead Auditor and QA Director or Manager determines the scope of each audit. The QA Director or QA Manager may initiate special audits or expand the scope of audits. The Lead Auditor directs the audit team in developing checklists, instructions, or plans and performing the audit. The audit shall be conducted in accordance with the checklists, but the scope may be expanded by the audit team during the audit. The audit team consists of one or more auditors.

Auditors and lead auditors are responsible for performing audits in accordance with the applicable QA procedures. Auditors and lead auditors hold certifications as required by the QA Program. Additional details can be found in Appendix A of this chapter. Before being certified under the LES QA Program, auditors must complete training on the following topics:

- LES QA Program
- Audit fundamentals, including audit scheduling, planning, performance, reporting, and follow-up action involved in conducting audits
- Objectives and techniques of performing audits
- On-the-job training.

Certification of auditors and lead auditors is based on the QA Director's or Manager's evaluation of education, experience, professional qualifications, leadership, sound judgment, maturity, analytical ability, tenacity, and past performance and completion of QA training courses. A lead auditor must also have participated in a minimum of five QA audits or audit equivalent within a period of time not to exceed three years prior to the date of certification. Audit equivalents include assessments, pre-award evaluations or comprehensive surveillances (provided the

prospective lead auditor took part in the planning, checklist development, performance, and reporting of the audit equivalent activities). One audit must be a nuclear-related QA audit or audit equivalent within the year prior to certification.

Personnel performing assessments do not require certification, but they are required to complete QA orientation training, as well as training on the assessment process. The nuclear criticality safety assessments are performed under the direction of the criticality safety staff. Personnel performing these assessments do not report to the production organization and have no direct responsibility for the function or area being assessed.

Appendix A, Section 18 “Audits” of this chapter provides additional details regarding the QA Audit program requirements.

(This page intentionally left blank)

11.6 INCIDENT INVESTIGATIONS AND CORRECTIVE ACTION PROCESS

The incident investigation and corrective action process described in this section is consistent with that previously submitted for NRC review in Section 11.4 and Section 10.16 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC Staff reviewed the previous submittal and found it to be acceptable. The NRC Staff's review and conclusions associated with the incident investigation and corrective action process are documented in Section 10.7.6 and Section 12 of NUREG-1491 (NRC, 1994).

11.6.1 Incident Investigations

The incident investigation process is a simple mechanism available for use by any person at the facility for reporting deficiencies, abnormal events and potentially unsafe conditions or activities. Abnormal events that potentially threaten or lessen the effectiveness of health, safety or environmental protection will be identified and reported to and investigated by the HS&E Manager. 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. 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 CAP procedures. Radiological, criticality, hazardous chemical, and industrial safety requirements shall be addressed. Guidance for classifying occurrences shall be contained in CAP procedures, including examples of threshold off-normal occurrences. 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 HS&E Manager is responsible for:

- Maintaining a list of agencies to be notified
- Determining if a report to an agency is required
- Notifying the agency when required.

The licensing organization has the responsibility for all appropriate communications with government agencies.

The HS&E Manager or designee shall maintain a record of corrective actions to be implemented as a result of off-normal occurrence investigations in accordance with CAP procedures. These corrective actions shall include documenting lessons learned, and implementing worker training where indicated, and shall be tracked to completion by the HS&E Manager or designee.

Specifics of the Incident Investigation process are as follows:

1. LES will establish a process to investigate abnormal events that may occur during operation of the facility, to determine their specific or generic root cause(s)and generic implications, to recommend corrective actions, and to report to the NRC as required by 10 CFR 70.50 (CFR, 2003c) and 70.74 (CFR, 2003f). The investigation process will include a prompt risk-based evaluation and, depending on the complexity and severity of

the event, one individual may suffice to conduct the evaluation. The investigator(s) will be independent from the line function(s) involved with the incident under investigation and are assured of no retaliation for participating in investigations. Investigations will begin within 48 hours of the abnormal event, or sooner, depending on safety significance of the event. The record of IROFS failures required by 10 CFR 70.62(a)(3) (CFR, 2003d) for IROFS will be reviewed as part of the investigation. Record revisions necessitated by post-failure investigation conclusions will be made within five working days of the completion of the investigation.

2. Qualified internal or external investigators are appointed to serve on investigating teams when required. The teams will include at least one process expert and at least one team member trained in root cause analysis.
3. LES will monitor and document corrective actions through completion.
4. LES will maintain auditable records and documentation related to abnormal events, investigations, and root cause analyses so that "lessons learned" may be applied to future operations of the facility. For each abnormal event, the incident report includes a description, contributing factors, a root cause analysis, findings, and recommendations. Relevant findings are reviewed with all affected personnel. Details of the event sequence will be compared with accident sequences already considered in the ISA, and the ISA Summary will be modified to include evaluation of the risk associated with accidents of the type actually experienced.

LES will develop CAP procedures for conducting an incident investigation, and the procedures will contain the following elements:

1. A documented plan for investigating an abnormal event.
2. A description of the functions, qualifications, and/or responsibilities of the manager who would lead the investigative team and those of the other team members; the scope of the team's authority and responsibilities; and assurance of cooperation of management.
3. Assurance of the team's authority to obtain all the information considered necessary and its independence from responsibility for or to the functional area involved in the incident under investigation.
4. Retention of documentation relating to abnormal events for two years or for the life of the operation, whichever is longer.
5. Guidance for personnel conducting the investigation on how to apply a reasonable, systematic, structured approach to determine the specific or generic root cause(s) and generic implications of the problem.
6. Requirements to make available original investigation reports to the NRC on request.
7. A system for monitoring the completion of appropriate corrective actions.

11.6.2 Corrective Action Process

The LES QA Program identifies the responsibilities and provides authority for those individuals involved in quality activities to identify any condition adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective materials and equipment, and non-

conformances. These individuals identify and document conditions adverse to quality, analyze and determine how the conditions can be corrected or resolved, and take such steps as necessary to implement corrective actions in accordance with documented procedures.

The QA Program requires regularly scheduled audits and assessments to ensure that needed corrective actions are identified. LES employees have the authority and responsibility to initiate the corrective action process if they discover deficiencies. The QA Program contains procedures for identifying, reporting, resolving, documenting, and analyzing conditions adverse to quality. Reports of conditions adverse to quality are analyzed to identify trends in quality performance. Significant conditions adverse to quality and significant trends are reported to senior management in accordance with CAP procedures.

Follow-up action is taken by the QA Manager to verify proper and timely implementation of corrective action.

Significant conditions adverse to quality, the cause of the conditions and the corrective action taken to preclude repetition are documented and reported to management for review and assessment in accordance with CAP procedures.

Appendix A, Section 16 “Corrective Action” of this chapter provides additional details regarding the CAP requirements.

11.7 RECORDS MANAGEMENT

The management measure described in this section is consistent with that previously submitted for NRC review in Section 11.4 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC Staff reviewed the previous submittal and found it to be acceptable. The NRC Staff's review and conclusions associated with records management are documented in Section 10.6 of NUREG-1491 (NRC, 1994).

Records management shall be performed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures. QA records are not considered valid until they are authenticated and dated by authorized personnel.

The LES QA Program requires procedures for reviewing, approving, handling, identifying, retention, retrieval and maintenance of quality assurance records. These records include the results of tests and inspections required by applicable codes and standards, construction, procurement and receiving records, personnel certification records, design calculations, purchase orders, specifications and amendments, procedures, incident investigation results and approvals or corrective action taken, various certification forms, source surveillance and audit reports, component data packages, and any other QA documentation required by specifications or procedures. These records are maintained at locations where they can be reviewed and audited to establish that the required quality has been assured.

For computer codes and computerized data used for activities relied on for safety, as specified in the ISA Summary, procedures are established for maintaining readability and usability of older codes and data as computing technology changes. For example, procedures allow older forms of information and codes for older computing equipment to be transferred to contemporary computing media and equipment.

The facility maintains a Master File that access to, and use of is controlled. Documents in the Master File shall be legible and shall be identifiable as to the subject to which they pertain. Documents shall be considered valid only if stamped, initialed, signed or otherwise authenticated and dated by authorized personnel. Documents in the Master File may be originals or reproduced copies. Computer storage of data may be used in the Master File.

In order to preclude deterioration of records in the Master File, the following requirements are 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, e.g., radiographs, photographs, negatives, microfilm, 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. Dual-facility records storage uses an electronic data management system and storage of backup tapes in a fireproof safe.

The Master File storage system shall provide for the accurate retrieval of information without undue delay. Written instructions shall be prepared regarding the storage of records in a Master

File, and a supervisor shall be designated the responsibility for implementing the requirements of the instructions. These instructions shall 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.

Records related to health and safety shall be maintained in accordance with the requirements of Title 10, Code of Federal Regulations. The following records shall be retained for at least the periods indicated in accordance with the Records Management procedures which specifies retention periods

The following are examples of records that will be retained:

- Operating logs
- Procedures
- Supplier QA documentation for equipment, materials, etc.
- Nonconforming item reports
- Test documentation/test results - preoperational/operational
- Facility modification records
- Drawings/specifications
- Procurement documents (e.g., purchase orders, purchase requisitions)
- Nuclear material control and accounting records
- Maintenance activities including calibration records
- Inspection documentation (plant processes)

- Audit reports
- Reportable occurrences and compliance records
- Completed work orders
- License conditions (specifications) records
- Software verification records
- System descriptions
- As-built design documentation packages
- Regulatory reports and corrective action.

Other retention times are specified for other facility records as necessary to meet applicable regulatory requirements. These retention times are indicated in facility administrative procedures.

Appendix A, Section 17 “Quality Assurance Records” of this chapter provides additional details regarding records management requirements.

11.8 OTHER QA ELEMENTS

The QA Program and its supporting manuals, procedures and instructions are applicable to items and activities designated as QA Level 1 and 2.

The QA Director is responsible for developing and revising the QA Program and assuring it is in compliance with applicable regulations, codes and standards. The QA Director approves the supporting manuals, procedures, and revisions for their respective scope of responsibility.

The QA Program specifies mandatory requirements for performing activities affecting quality and is set forth in procedures which are distributed on a controlled basis to organizations and individuals responsible for quality. Revisions to these procedures are also distributed on a controlled basis. Applicable portions of the QA Program are documented, approved and implemented prior to undertaking an activity.

A management assessment of the QA program is performed at least six months prior to scheduled receipt of licensed material on the site. Items identified as needing completion or modification are entered into the CAP and corrective action completed before scheduled receipt of licensed material. LES Management monitors the QA program prior to this initial management assessment through project review meetings and annual assessments. This management assessment along with integrated schedules and program review meetings ensure that the QA program is in place and effective prior to receiving licensed material.

The LES QA program for design, construction, and preoperational testing continues simultaneously with the QA program for the operational phase while construction activities are in progress.

Anyone may propose changes to the QA Program supporting manuals and procedures. When reviewed by the QA Director and found acceptable and compatible with applicable requirements, guidelines and LES policy, the changes may be implemented. The QA Program and supporting manuals and procedures are reviewed periodically to ensure they are in compliance with applicable regulations, codes, and standards. New or revised regulations, codes, and standards are reviewed for incorporation into the QA Program and supporting manuals and procedures as necessary.

Personnel performing activities covered by the QA program shall perform work in accordance with approved procedures, and must demonstrate suitable proficiency in their assigned tasks. Formal training programs are established for quality assurance policies, requirements, procedures, and methods. Ongoing training is provided to ensure continuing proficiency as procedural requirements change. New employees are required to attend a QA indoctrination class on authority, organization, policies, manuals, and procedures.

Additional formal training is conducted in specific topics such as NRC regulations and guidance, procedures, auditing, and applicable codes and standards. Supplemental training is performed as required. On-the-job training is performed by the employee's supervisor in QA area-specific procedures and requirements. Training records are maintained for each person performing quality-related job functions.

The LES President assesses the scope, status, adequacy and regulatory compliance of the QA Program through regular meetings and correspondence with the Plant Manager and the LES

QA organization. Additionally, LES QA, through the QA Director, periodically informs the LES President and Plant Manager of quality concerns that need management resolution.

LES participates in the planning and scheduling for system turnover as construction is completed. Prior to system turnover, written procedures are developed for control of the transfer of systems, structures, components and associated documentation. The procedures include checklists, marked drawings, documentation lists, system status, and receipt control.

Major work activities contracted by LES shall be identified and controlled. Principal contractors shall be required to comply with the applicable portions of 10 CFR 50, Appendix B (CFR, 2003b), as determined by LES. The performance of contracted activities shall be formally evaluated by LES commensurate with the importance of the activities to safety.

Facility components and processes are assigned a QA level based on their safety significance. Each component will receive a classification of QA Level 1, QA Level 2, or QA Level 3 that applies throughout the life of the facility and is based on the following definitions:

QA Level 1 Requirements

The QA Level 1 Program shall conform to the criteria established in 10 CFR 50, Appendix B (CFR, 2003b). These criteria shall be met by commitments to follow the guidelines of ASME NQA-1-1994 (ASME, 1994), including supplements as revised by the ASME NQA-1a-1995 Addenda (ASME, 1995) as specified in the QA Program Description. The QA Level 1 QA program shall be applied to those structures, systems, components, and administrative controls that have been determined to be IROFS, items that affect the functions of the IROFS, and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied.

QA Level 2 Requirements

The QA Level 2 program is an owner-defined QA program that uses the ASME NQA-1-1994 standard (ASME, 1994), including supplements as revised by the ASME NQA-1a-1995 Addenda (ASME, 1995) as guidance. General QA Level 2 requirements are described in Section 20, “Quality Assurance Program for QA Level 2 Activities”. For contractors, the QA Level 2 program shall be described in documents that must be approved by LES. The QA Level 2 program shall be applied to Owner designated structures, systems, components, and activities. An International Organization for Standardization (ISO) 9000 series QA program may be acceptable for QA Level 2 applications provided it complies with LES Quality Assurance Program Description requirements. The QA program manual must be reviewed and accepted by the LES QA Director.

QA Level 3 Requirements

The QA Level 3 program is defined as standard commercial practice. A documented QA Level 3 program is not required. QA Level 3 governs all activities not designated as QA Level 1 or QA Level 2.

Appendix A, “LES Quality Assurance Program Description” of this chapter provides additional details and commitments to other QA elements that will be implemented to support the Management Measures described in this chapter.

11.9 REFERENCES

ANSI, 1991. Nuclear Criticality Safety Training, ANSI/ANS-8.20-1991, American National Standards Institute/American Nuclear Society, 1991.

ANSI, 1994. Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, ANSI/ANS-3.2-1994, American National Standards Institute/American Nuclear Society, 1994.

ANSI, 1996. Administrative Practices for Nuclear Criticality Safety, ANSI/ANS-8.19-1996, American National Standards Institute/American Nuclear Society, 1996.

ASME, 1994. Quality Assurance Requirements for Nuclear Facility Applications, ASME NQA-1-1994, American Society of Mechanical Engineers, 1994.

ASME, 1995. Addenda to ASME NQA-1-1994 Edition, Quality Assurance Requirements for Nuclear Facility Applications, ASME NQA-1a-1995, American Society of Mechanical Engineers, 1995.

CFR, 1994. Title 10, Code of Federal Regulations, Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants, 1994.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 21, Reporting of Defects and Noncompliance, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.50, Reporting requirements, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.62, Safety program and integrated safety analysis, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.72, Facility changes and change process, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Section 70.74, Additional reporting requirements, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

LES, 1993. Claiborne Enrichment Center Safety Analysis Report, Louisiana Energy Services, December 1993.

NRC, 1994. Safety Evaluation Report for the Claiborne Enrichment Center, Homer, Louisiana, NUREG-1491, U.S. Nuclear Regulatory Commission, January 1994.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

NRC, 2004. Safety Evaluation Report: Louisiana Energy Services Quality Assurance Program Description for the National Enrichment Facility, U.S. Nuclear Regulatory Commission, April 9, 2004.

APPENDIX A

Louisiana Energy Services

Quality Assurance Program Description

Design, Construction, Operations and Decommissioning Phases

TABLE OF CONTENTS

		Page
INTRODUCTION		A1
SECTION 1	ORGANIZATION	A3
SECTION 2	QUALITY ASSURANCE PROGRAM	A9
SECTION 3	DESIGN CONTROL	A13
SECTION 4	PROCUREMENT DOCUMENT CONTROL	A21
SECTION 5	INSTRUCTIONS, PROCEDURES, AND DRAWINGS	A23
SECTION 6	DOCUMENT CONTROL	A25
SECTION 7	CONTROL OF PURCHASED MATERIAL, EQUIPMENT AND SERVICES	A27
SECTION 8	IDENTIFICATION AND CONTROL MATERIALS, PARTS AND COMPONENTS	A33
SECTION 9	CONTROL OF SPECIAL PROCESSES	A35
SECTION 10	INSPECTION	A37
SECTION 11	TEST CONTROL	A41
SECTION 12	CONTROL OF MEASURING AND TEST EQUIPMENT	A45
SECTION 13	HANDLING, STORAGE, AND SHIPPING	A47
SECTION 14	INSPECTION, TEST, AND OPERATING STATUS	A49
SECTION 15	NONCONFORMING ITEMS	A51
SECTION 16	CORRECTIVE ACTION	A53
SECTION 17	QUALITY ASSURANCE RECORDS	A55
SECTION 18	AUDITS	A61
SECTION 19	PROVISIONS FOR CHANGE	A65
SECTION 20	QUALITY ASSURANCE PROGRAM FOR QA LEVEL 2 ACTIVITIES	A67

LIST OF FIGURES

Figure A1 LES Corporate, Design and Construction Organization
Figure A2 LES National Enrichment Facility Operating Organization

INTRODUCTION

Louisiana Energy Services (LES) maintains full responsibility for ensuring that the enrichment facility is designed, constructed, operated, and decommissioned in conformance with applicable regulatory requirements, specified design requirements, applicable industry standards and good engineering practices in a manner to protect the health and safety of the employees and the public. To this end, the LES Quality Assurance Program conforms to the criteria established in Title 10 of the Code of Federal Regulations 10 CFR 50, Appendix B, Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants. The criteria in 10 CFR 50, Appendix B, are met by LES's commitment to follow the guidelines of the American Society of Mechanical Engineers (ASME) Quality Assurance (QA) standard NQA-1-1994, Quality Assurance Program Requirements for Nuclear Facilities, including supplements as revised by the ASME NQA-1a-1995 Addenda.

The LES QA Program described herein covers design, construction (including pre-operational testing), operation (including testing), maintenance and modification, and decommissioning of the facility. This Quality Assurance Program Description (QAPD) describes the requirements to be applied to those structures, systems and components, and activities that have been designated Quality Assurance (QA) Level 1.

QA Level 1 is applied exclusively to items relied on for safety (IROFS), any items which are determined to affect the function of the IROFS, and, in general, to items required to satisfy regulatory requirements. The development of the IROFS list is a product of the Integrated Safety Analysis (ISA) process. Chapter 3, Integrated Safety Analysis Summary of the LES Safety Analysis Report (SAR) provides the methodology utilized to establish the IROFS list. IROFS are comprised of specific structures, systems and components (SSC) and administrative controls. All sections of this QAPD are applied to IROFS, any SSC and administrative controls which are determined to affect the functions of the IROFS and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied. Application of the QAPD requirements is part of the configuration management system and will be performed in accordance with documented procedures. The LES QA organization reviews and concurs with the selection of the IROFS and the application of QA requirements to the IROFS, any items which are determined to affect the functions of the IROFS and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied.

The QA Level 2 program description is provided in Section 20, Quality Assurance Program for QA Level 2 Activities of this QAPD. These requirements are implemented by LES and LES contractors through the use of approved QA programs and procedures. The Owner defined QA Level 2 SSCs and their associated activities i.e., those SSCs that are not IROFS, provide support of normal operations of the facility, and do not affect the functions of the IROFS (e.g., occupational exposure, radioactive waste management) and SSCs that minimize public, worker, and environmental risks (e.g., physical interaction protection, certain radiation monitors and criticality alarms) are evaluated against the requirements in Section 20, of this QAPD. This evaluation identifies which QA controls are needed to ensure these SSC meet their intended functions and do not affect the functions of the IROFS. This evaluation may also include nuclear industry precedent in the application of augmented QA requirements.

Three QA Levels have been established and apply throughout the life of the facility from licensing and siting through design, construction, operation, and decommissioning. The three

levels are defined as follows.

QA LEVEL 1 REQUIREMENTS

The QA Level 1 Program shall conform to the criteria established in 10 CFR 50, Appendix B. These criteria shall be met by commitments to follow the guidelines of ASME NQA-1-1994, including supplements as revised by the ASME NQA-1a-1995 Addenda. The QA Level 1 QA program shall be applied to those structures, systems, components, and administrative controls that have been determined to be IROFS, items that affect the functions of the IROFS, and, in general, to items required to satisfy regulatory requirements.

QA LEVEL 2 REQUIREMENTS

The QA Level 2 program is an owner-defined QA program that uses the ASME NQA-1 standard as guidance. General QA Level 2 requirements are described in Section 20, Quality Assurance Program for QA Level 2 Activities. For contractors, the QA Level 2 program shall be described in documents that must be approved by LES. The QA Level 2 program shall be applied to Owner designated structures, systems, components, and activities. An International Organization for Standardization (ISO) 9000 series QA program may be acceptable for QA Level 2 applications provided it complies with applicable LES QAPD requirements and the QAPD is reviewed and accepted by the LES QA Director.

QA LEVEL 3 REQUIREMENTS

The QA Level 3 program is defined as standard commercial practice. A documented QA Level 3 program is not required. QA Level 3 governs all activities not designated as QA Level 1 or QA Level 2.

As described in Section 19, Provisions for Change, subsequent changes to the LES QA Program shall be incorporated in this QAPD. Any changes that reduce the commitments in the approved QAPD will be submitted to the Nuclear Regulatory Commission (NRC) for review and approval prior to implementation.

SECTION 1 ORGANIZATION

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 1, Organization, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 1 and Supplement 1S-1 of NQA-1-1994.

LES employees and contractor employees representing LES have full responsibility to ensure that the facility is designed, constructed, operated, and decommissioned in a manner to protect the health and safety of the public. This responsibility begins with initial design and continues throughout the life of the facility. The LES QA Program is designed to ensure that the necessary quality requirements for structures, systems, components and work activities are met. This objective is attained by ensuring that the organizational structure and the responsibility assignments are such that (a) quality is achieved and maintained by those who have been assigned responsibility for performing work and, (b) quality achievement is verified by persons or organizations not directly responsible for performing the work.

CORPORATE ORGANIZATION AND FUNCTIONS

LES is the owner and operator of the enrichment facility. LES is a registered limited partnership formed to provide uranium enrichment services for commercial nuclear power plants. LES is responsible for the design, construction, operation and decommissioning of the enrichment facility in accordance with its QA Program. The President of LES reports to the LES Management Committee. The committee is composed of representatives from the general partners of LES.

The LES President establishes the basic policies of the QA Program. These policies are described in this QA Program, are transmitted to all levels of management, and are implemented through approved procedures. The LES QA Director has overall responsibility for development, management and implementation of the LES QA Program during all phases of the enrichment facility. As part of this responsibility, the QA Director is responsible for ensuring that contractor QA Programs meet all applicable requirements of the LES QA Program. LES management is continually involved in activities affecting quality and QA requirements.

Reporting to the President are the Engineering and Contracts Manager, Corporate Communications Manager, Chief Financial Officer (CFO), Quality Assurance Director, Chief Operating Officer (COO) and the Health, Safety and Environment Manager. Figure A1, LES Corporate, Design and Construction Organization, shows the levels of authority and lines of communications for activities affecting quality.

DESIGN AND CONSTRUCTION ORGANIZATION AND FUNCTIONS

The LES Engineering and Contracts Manager or the LES President acting in the capacity of the Engineering and Contracts Manager, has contracted Urenco, the owner of the enrichment technology and operator of enrichment facilities in Europe, to prepare the reference design for the facility. An architect/engineering (A/E) firm has been contracted and is under the responsibility of the Engineering and Contracts Manager or President to further specify structures and systems of the facility, and ensure the reference design meets all applicable U.S. codes and standards. A contractor specializing in site evaluations has been contracted and is under the responsibility of the Engineering and Contracts Manager or President to perform the site selection evaluation. A nuclear consulting company has been contracted and is under the

responsibility of the Engineering and Contracts Manager or President to conduct the site characterization, perform the Integrated Safety Analysis and to support development of the license application including the Environmental Report.

During the design and construction phases, preparation of design and construction documents and construction itself are contracted to qualified contractors. The Engineering and Contracts Manager is responsible for managing the design, construction and construction inspection activities, startup, including pre-operational testing and procurement activities during these phases. Contractor QA Programs will be reviewed by the LES QA organization and must be approved by the LES QA Director before work can start as described in Section 4, Procurement Document Control, and Section 7, Control of Purchased Material, Equipment and Services. Urenco will design, manufacture and deliver to the site the centrifuges necessary for the facility under a QA Program approved by the LES QA Director or under the LES QA Program. In addition, Urenco is supplying the technical assistance and consultation for the facility in accordance with the applicable requirements of the LES QA Program. As shown in Figure A1, the Engineering and Contracts Manager is responsible for managing the work and contracts with the Technology Supplier (i.e.,Urenco) and a select group of Project Managers. These Project Managers will be responsible for the areas of Procurement, Construction, Engineering, Project Engineering, Project Controls and Start up.

QA Procedures will be developed by the Engineering and Contracts organization to implement this QAPD in the Engineering and Contracts area.

OPERATING ORGANIZATION AND FUNCTIONS

The operating organization is shown in Figure A2, LES National Enrichment Facility Operating Organization. The Plant Manager reports to the COO and is responsible for the overall operation and administration of the enrichment facility. The Plant Manager is also responsible for ensuring the facility complies with all applicable regulatory requirements including the requirements of this QAPD. In the discharge of these responsibilities, the Plant Manager directs the activities of the following groups.

- Health, Safety and Environment
- Operations
- Uranium Management
- Technical Services
- Human Resources
- Quality Assurance

Procedures will be developed by the respective operations organizations to implement the requirements of this QAPD. Specific details of organizational responsibilities and job descriptions are provided in the National Enrichment Facility (NEF) Safety Analysis Report.

QA ORGANIZATION AND FUNCTIONS

The LES QA organization during the design and construction phases will be headed by the LES QA Director. The LES QA Director reports directly to the LES President and is vested with the authority, access to work areas, and organizational independence to ensure that the requirements of this QAPD are properly implemented.

The LES QA Director is responsible for managing the LES QA Program that includes the following activities:

- QA Technical Support
  - Maintain the LES QAPD
  - Maintain QA procedures
  - QA technical reviews of procurement documents
  - Review and concurrence of changes to the identified IROFS, items that could affect the functions of IROFS, and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied
  - Administer the Corrective Action and Nonconformance Processes
  - Maintain the LES Approved Suppliers List (ASL)
  - Administer the Auditor and Lead Auditor Certification Process
  - QA reviews of project documents
  - Approval of contractor QA Programs
  - Oversight of contractor QA Programs Implementation
  - Oversight of the quality of design and construction, including but not limited to the ISA process and the resultant selection of IROFS
  - Oversight of document and records control
- QA Verification
  - Audits, surveillances and assessments
  - Contractor/supplier evaluations
  - Contractor nonconformances
  - Equipment/Vendor Shop Inspections
  - Witness vendor acceptance testing

During the transition from construction to operations, when startup testing and plant operations may be concurrent as the facility is completed in phases, a plant QA Manager will be added to the LES QA Organization. During this transition period as well as during operations, the plant QA Manager will report to the Plant Manager. However, the plant QA Manager has the authority and responsibility to contact the LES President, through the QA Director, with any QA concerns during startup and plant operations. After construction has been completed on the facility the corporate functions reporting the LES QA Director, i.e., QA Technical Support and QA Verification; will transition to the plant QA Manager. During the operations and decommissioning phases, the LES QA Director will advise the LES President on quality-related matters and continue to have governance and oversight responsibilities with respect to the QA organization headed by the plant QA Manager. The following additional QA Manager responsibilities are included for start up testing and operations:

- QA Technical Support
  - Quality Engineering support of startup organization
  - Oversight of startup activities

- QA selected reviews and oversight of programs developed for operations, including but not limited to the ISA process, the identification of IROFS and items that affect the performance of IROFS and any changes thereto, the controls for assuring IROFS performance and verifying and maintaining the facility design basis.
- QA selected reviews and oversight of operations including maintenance and testing and modification procedures
- Review and concurrence of changes to the identified IROFS, items that could affect the functions of IROFS, and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied
- QA Oversight of operations procedure implementation
- Quality Control (QC) Inspection certification process

- QC Inspections
  - Receipt Inspections of QA Level 1 items
  - Applicable discipline inspections of modifications to QA Level 1 components

Accordingly, during the transition from construction to operations, the operations phase, and the decommissioning phase, the management of the QA organization and the QA staff have the responsibility to make quality assurance decisions and have sufficient authority, access to work areas, and organizational freedom to:

- Identify quality problems
- Initiate and recommend solutions to quality problems through designated channels
- Verify implementation of solutions
- Assure that further processing, delivery, installation, or use of items is controlled until proper disposition of nonconformances, deficiencies or unsatisfactory conditions has occurred
- Have direct access to highest levels of management
- Be sufficiently independent from cost and schedule considerations and have stop-work authority.

ORGANIZATIONAL INTERFACES

The organizational interfaces between LES, contractors, and project applicable regulatory agencies are identified in the appropriate plans, contracts and implementing procedures. These documents contain the appropriate protocols, applicable roles, responsibilities and approval authorities for the specific topics for which they apply. LES design interfaces shall be identified and procedurally controlled. Design efforts shall be coordinated among interfacing organizations as detailed in LES procedures. Interface controls shall include the assignment of responsibility and the establishment of implementing documents among interfacing design organizations for the review, approval, release, distribution and revision of documents involving design interfaces. LES design information transmitted across interfaces shall be documented and procedurally controlled. LES transmittals of design information and/or documents shall reflect the status of the transmitted information and documents. Incomplete designs that require further evaluation, review or approval shall be identified. When it is necessary to initially transmit the design information orally or by other informal means, design information shall be promptly confirmed through a controlled implementing document.

DELEGATION OF WORK

The delegation of work between LES and contractors is identified in applicable plans, contracts and implementing procedures. In all cases of delegation, LES retains the overall responsibility for all work performed under the direction of LES. All LES QA Level 1 work activities shall meet the requirements of this QAPD. Responsible managers have the authority to delegate tasks to another qualified individual within their organization provided the designated individual possesses the required qualifications and these qualifications are documented. All delegations shall be in writing. The responsible manager retains the ultimate responsibility and accountability for implementing the applicable requirements.

RESOLUTION OF DISPUTES

Disputes involving differences of opinion on quality matters or issues are brought to the attention of line management, and if not resolved by the individual's manager, are elevated progressively to the QA Director. If satisfactory resolution cannot be obtained at that level, the matter is then elevated to the LES President for final resolution.

WORKER RESPONSIBILITIES

Each employee has an obligation to identify concerns using the corrective action process with respect to work within their scope of responsibility whenever the health and safety of our workers, the public, or the environment is involved or when continued work will produce results that are not in compliance with the LES QA Program. This process is controlled by an LES procedure, which applies across the entire project/facility. The authorities and responsibilities for stopping work, the criteria and documentation required to process the stop work and the actions required before work may resume are detailed in an LES procedure. This process ensures that safety related activities are controlled until the deficiency, or unsatisfactory condition, has been resolved. Worker responsibilities are further discussed in Section 16, Corrective Action.

SECTION 2 QUALITY ASSURANCE PROGRAM

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 2, Quality Assurance Program, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 2 and Supplements 2S-1, 2S-2, 2S-3 and 2S-4 of NQA-1-1994 Part I as revised by NQA-1a-1995 Addenda of NQA-1-1994.

PROGRAM BASIS

The LES Quality Assurance Program complies with 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, and applies to all levels of the organization, including contractors, who perform QA Level 1 activities. Part I and selected sections of Part II of ASME NQA-1-1994, Quality Assurance Requirements for Nuclear Facility Applications, as revised by NQA-1a-1995 Addenda are used in conjunction with 10 CFR 50, Appendix B and provide additional detailed quality assurance guidelines which are committed to in this QAPD. The LES QAPD describes LES's overall compliance with 10 CFR 50, Appendix B and commitments to ASME NQA-1. This document states LES policies, assigns responsibilities and specifies requirements governing implementation of the QA Program to the design, construction, operation and decommissioning of the LES enrichment facility. All 18 criteria of 10 CFR 50, Appendix B have been addressed to identify the scope of QA Program applied to the LES enrichment facility. QA requirements will also apply to contractors as delineated in procurement documents controlled under Section 4, Procurement Document Control, of this QAPD. The necessary management measures to control the quality of subcontracted activities for the LES design, procurement, and installation and testing of QA Level 1 components and activities have been established in this QAPD. The QAPD will be reviewed for needed revisions as described in Section 19, Provisions For Change.

Specific processes and controls, which implement the provisions of 10 CFR 50, Appendix B and the commitment to ASME NQA-1-1994, as specified in this QAPD are delineated in procedures. Development, review, approval and training on procedures shall be performed prior to performance of the activities controlled by the procedures.

The QA Program provides for the planning and accomplishment of activities affecting quality under suitably controlled conditions. Controlled conditions include the use of appropriate equipment, suitable environmental conditions for accomplishing the activity, and assurance that prerequisites for the given activity have been satisfied. The LES QA Program provides for special controls, processes, test equipment, tools and skills to attain the required quality and verification of quality. QA requirements contained in this QAPD are also invoked on LES contractors for their contracted scope of work.

When work cannot be accomplished as specified in implementing QA procedures, or accomplishment of such work would result in an adverse condition, work is stopped until proper corrective action is taken. If procedures cannot be used as written, then work is stopped until the procedures are changed. Requirements for stop work are further discussed in Section 16, Corrective Action.

Flowdown of QA Requirements to Contractors and Suppliers

QA requirements for QA Level 1 activities are imposed on LES contractors and suppliers through the respective procurement documents for the particular scope of work being

contracted. Determination of the specific QA requirements, supplier evaluations, and proposal/bid evaluations are in accordance with the requirements of Section 4, Procurement Document Control, and Section 7, Control of Purchased Material, Equipment and Services, of this document. Applicable QA Program elements required for the particular scope of work are identified in procurement documents. Potential contractors/suppliers are required to submit their QA Programs to the LES QA organization for review in accordance with the request for proposal/procurement specification. The LES QA organization performs an audit at the contractor's/supplier's facility of their QA program and its implementation verifying that the contractor's/supplier's QA program meets the requirements established in the request for proposal/procurement specification. If the audit is acceptable then the contractor/supplier is added to the LES ASL and a contract between LES and the contractor/supplier may be issued. For procured items, LES may also require that the LES QA organization perform source inspections or witness tests at the supplier's facility prior to shipment if the equipment/component warrants inspection due to its safety significance and/or complexity. Such requirements are also identified in the procurement documents and/or contract.

Construction contractors for LES QA Program controlled construction activities are required to be placed on the ASL prior to contract award. Construction contractors are required to perform the QA activities required by their QA program including audits of their own activities as well as any required quality control (QC) inspections. The LES QA organization will provide oversight of these contractors in the form of audits and surveillances verifying that each contractor is properly implementing its QA program as approved by LES QA. Contractually contractors will be required to promptly correct LES identified deficiencies and noncomformances.

IDENTIFICATION AND APPLICATION OF QA CONTROLS

QA Level 1 is applied exclusively to IROFS, any items which are determined to affect the function of the IROFS, and, in general, to items required to satisfy regulatory requirements. Since the development of the IROFS list is a product of the ISA process, the applicable QA Level 1 requirements are also applied to this process. Chapter 3, Integrated Safety Analysis Summary of the NEF Safety Analysis Report (SAR) provides the methodology utilized to establish the IROFS list. IROFS are comprised of specific structures, systems and components (SSC) and administrative controls. All applicable sections of this QAPD are applied to IROFS, any SSC and administrative controls which are determined to affect the functions of the IROFS and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied. Application of the QAPD requirements is part of the configuration management program used to verify and maintain the facility design basis and will be performed in accordance with documented procedures. Accordingly, as described in Section 1, Organization, the QA organization is responsible for selected reviews and oversight of these processes and programs. In particular, the LES QA organization reviews and concurs with the selection of the IROFS and the application of QA requirements to the IROFS, any items which are determined to affect the functions of the IROFS and items required to satisfy regulatory reqirements for which QA Level 1 requirements are applied.

The QA Level 2 program description is provided in Section 20, Quality Assurance Program for QA Level 2 Activities of this QAPD. These requirements are implemented by LES and LES contractors through the use of approved QA programs and procedures. The Owner defined QA Level 2 SSCs and their associated activities i.e., those SSCs that are not IROFS, provide support of normal operations of the facility, and do not affect the functions of the IROFS (e.g., occupational exposure, radioactive waste management) and SSCs that minimize public, worker,

and environmental risks (e.g., physical interaction protection, certain radiation monitors and criticality alarms) are evaluated against the requirements in Section 20, of this QAPD. This evaluation identifies which QA controls are needed to ensure these SSCs meet their intended functions and do not affect the functions of the IROFS. This evaluation may also include nuclear industry precedent in the application of augmented QA requirements.

Three QA Levels have been established and apply throughout the life of the facility from licensing and siting through design, construction, testing, startup, operation, maintenance, modification, and decommissioning. The three levels are defined as follows.

QA LEVEL 1 REQUIREMENTS

The QA Level 1 Program shall conform to the criteria established in 10 CFR 50, Appendix B. These criteria shall be met by commitments to follow the guidelines of ASME NQA-1-1994, including supplements as revised by the ASME NQA-1a-1995 Addenda. The QA Level 1 QA program shall be applied to those structures, systems, components, and administrative controls that have been determined to be IROFS, items that affect the functions of the IROFS, and, in general, to items required to satisfy regulatory requirements.

QA LEVEL 2 REQUIREMENTS

The QA Level 2 program is an owner-defined QA program that uses the ASME NQA-1 standard as guidance. General QA Level 2 requirements are described in Section 20, Quality Assurance Program for QA Level 2 Activities. For contractors, the QA Level 2 program shall be described in documents that must be approved by LES. The QA Level 2 program shall be applied to Owner designated structures, systems, components, and activities. An International Organization for Standardization (ISO) 9000 series QA program may be acceptable for QA Level 2 applications provided it complies with LES QAPD requirements and the QAPD is reviewed and accepted by the LES QA Director.

QA LEVEL 3 REQUIREMENTS

The QA Level 3 program is defined as standard commercial practice. A documented QA Level 3 program is not required. QA Level 3 governs all activities not designated as QA Level 1 or QA Level 2.

QUALITY ASSURANCE TRAINING

LES employees who perform QA Level 1 activities receive LES QA Indoctrination Training. This training includes general criteria, including introduction to applicable codes, standards, QA Procedures, QA Program elements and job responsibilities and authorities. LES personnel assigned to perform QA Level 1 activities are also required to complete training in the specific LES QA procedures needed to perform their job roles and responsibilities as assigned by their supervisor. Detailed QA training is provided on the LES QA Program and job specific QA procedures prior to an employee beginning QA Level 1 work. Supervision is responsible for ensuring that personnel performing work under their supervision are appropriately trained. LES will also include a version of QA Indoctrination Training as part of the general employee training given to all full-time employees.

The Human Resources Manager is responsible for coordinating QA training activities for LES. Human Resources serves as a centralized training support service for supervision in coordinating training and maintaining QA training records. This responsibility is carried out as support for line management. LES supervisory personnel are responsible for determining the

type and extent of the training to be provided to an individual, and ensuring that the training is properly documented for personnel performing QA Level 1 activities. Retraining, when applicable, shall occur in order to maintain proficiency or when changes to work methods, technology, or job responsibilities occur. Such retraining is also documented.

MANAGEMENT ASSESSMENTS

The LES President is responsibility for ensuring that management assessments are conducted annually to determine if the LES QA Program is effective. Recommendations are provided to the LES President for action. Functional Managers and the QA Director conduct assessments annually of QA activities under their control. The managers report the results to the LES President for review. The results of these assessments are reviewed by senior management to determine the adequacy of implementation of the LES QA Program and to direct any needed changes for program improvements.

QUALIFICATION/CERTIFICATION OF INSPECTION AND TEST PERSONNEL

Inspection and test personnel performing QA Level 1 activities shall be certified in accordance with NQA-1-1994 Part I Supplement 2S-1, *Supplementary Requirements for the Qualification of Inspection and Test Personnel.*

QUALIFICATION/CERTIFICATION OF NONDESTRUCTIVE EXAMINATION (NDE) PERSONNEL

Nondestructive Examination (NDE) personnel performing QA Level 1 activities shall be certified in accordance with NQA-1a-1995 Part 1 Supplement 2S-2, *Supplementary Requirements for the Qualification of Nondestructive Examination Personnel and American Society of Nondestructive Testing (ASNT) Recommended Practice No. SNT-TC-1A, Personnel Qualification and Certification in Nondestructive Testing*, December 1988 Edition. Qualification/certification records meeting the requirements of Supplement 2S-2 shall be established and maintained as QA records.

QUALITY ASSURANCE AUDIT PERSONNEL

Audit personnel performing QA Level 1 activities shall be certified in accordance with NQA-1a-1995 Part 1 Supplement 2S-3 *Supplemental Requirements for the Qualification of Quality Assurance Program Audit Personnel*.

QUALITY ASSURANCE PROGRAM STATUS REPORTING TO MANAGEMENT

Management is regularly informed by the LES QA organization of adverse trends and lessons learned as a result of reviews conducted on audit reports, surveillance reports, corrective action reports, management assessments, etc. Corrective action is initiated as necessary.

SECTION 3 DESIGN CONTROL

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 3, Design Control, of 10 CFR 50, Appendix B, and the commitment to Basic Requirements 3 and Supplement 3S-1 of NQA-1-1994 Part I as revised by NQA-1a-1995 Addenda of NQA-1-1994. The LES QA Program also implements the commitment to Part II of NQA-1-1994 Subpart Part 2.7, *Quality Assurance Requirements of Computer Software for Nuclear Facility Applications,* as revised by NQA-1a-1995 Addenda of NQA-1-1994. These commitments also apply to computer software that is used to produce or manipulate data that is used directly in the design, analysis and operation of structures, systems and components relied on for safety. Part I, Supplement 11S-2, *Supplementary Requirements for Computer Program Testing*, requirements for computer software qualification and use are also implemented by the LES QA Program.

Measures are established in procedures to assure that applicable requirements are correctly translated into design documents. Design inputs are specified on a timely basis to support LES milestones. Controls are established for the selection and suitability of application of materials, parts, equipment and processes that are essential to the functions of structures, systems and components. Design interfaces to ensure completeness and efficiency of design are established in applicable procedures. Procedures detail the controls for design input, design process, design verification, design changes and approval. These procedures include appropriate quantitative and/or qualitative acceptance criteria for determining that activities have been satisfactorily accomplished. LES design documents are prepared, reviewed, and approved by qualified individuals. Design is verified by one or more of the following verification methods: design reviews, alternate calculations or qualification tests. Design changes are governed by control measures commensurate with those applied to the original design. The design process and design verification practices and procedures shall be reviewed and modified, as necessary, when a significant design change is required because of an incorrect design. These and any other design deficiencies discovered during the design process on subsequent design related activities that affect the design of SSC shall be entered into the Corrective Action Program (CAP) according to Section 16, Corrective Action. If these deficiencies cause constructed or partially constructed items (systems, structures or components) to be deficient, the affected items shall be controlled in accordance with Section 15, Nonconforming Items. Configuration management is maintained in accordance with the applicable procedure and the applicable procedures controlling changes to the various types of design documents.

DESIGN INPUT CONTROL

Applicable design inputs (such as design basis, conceptual design reports, performance requirements, regulatory requirements, codes and standards) shall be controlled by the LES Engineering and Contracts Manager according to the following requirements:

- Design inputs shall be identified and documented, and their selection reviewed and approved.
- Design inputs shall be specified and approved in a manner to support the schedule. Design inputs shall provide the necessary details to permit design to be carried out in a manner that provides a consistent basis for making design decisions, accomplishing design verification

and evaluating design changes.

- Changes from approved design inputs and reasons for the changes shall be identified, approved, documented and controlled.
- Design inputs based on assumptions that require re-verification shall be identified and controlled by the appropriate procedures.

DESIGN PROCESS

The LES design process shall be controlled by the Engineering and Contracts Manager according to the following requirements:

- LES design work shall be prescribed and documented on a timely basis and to the level of detail necessary to permit the design process to be carried out in a correct manner and to permit verification that the design meets requirements.
- Design documents shall be adequate to support design, construction and operation.
- Appropriate quality standards shall be identified and documented, and their selection reviewed and approved.
- Changes from specified standards, including the reasons for the change, shall be identified, approved, documented and controlled.
- Design methods, materials, parts, equipment and processes that are essential to the function of the structure, system, or component shall be selected and reviewed for and suitability of application.
- Applicable information derived from experience as set forth in reports or other documentation, shall be made available to cognizant design personnel.
- Final design documents (i.e., approved design output documents and approved changes thereto) shall be sufficiently detailed as to purpose, method, assumptions, design input, references and units such that a person technically qualified in the subject/engineering discipline can understand the documents and verify their adequacy without recourse to the originator of the design document.
- Procedural controls for identifying sub-assemblies or components on final design documents that are part of the item being designed shall be established. When a commercial grade item is modified and/or tested to new requirements that are different from the supplier's published product description, the component part shall be traceable to documentation noting that it is different from the originally approved commercial grade item.
- LES design drawings, specifications or other design output documents shall contain appropriate inspection, examination and testing acceptance criteria.

<u>DESIGN ANALYSIS</u>

LES design analyses shall be planned, controlled and documented. Design analysis documents shall be legible, in a form suitable for reproduction, filing and retrieval, and under configuration management control. LES design calculations shall be identifiable by subject (including structure, system or component to which the calculation applies), originator, reviewer and date, or by other designators in order that approved calculations are retrievable.

Computer software used to perform design analyses shall be developed and/or qualified, and used according to the provisions of ASME NQA-1-1994, Part II, Subpart 2.7 as revised by NQA-

1a-1995 Addenda and Supplement 11S-2. Computer software developed and/or qualified under the LES or its contractor QA programs may also be used to perform design analyses for LES, provided that the LES QA organization confirms these contractor QA programs meet the provisions NQA-1-1994, Part I, Supplement 11S-2 and NQA-1-1994 Part II, Subpart 2.7 as revised by NQA-1a-1995 addenda.

Computer programs may be utilized for design analysis without individual verification of the program for each application provided:

- The computer program has been verified to show that it produces correct solutions for the encoded mathematical model within defined limits for each parameter employed; and
- The encoded mathematical model has been shown to produce a valid solution to the physical problem associated with the particular application.

Computer programs shall be controlled to assure that changes are documented and approved by authorized personnel. Where changes to previously verified computer programs are made, verification shall be required for the change, including evaluation of the effects of these changes on the above.

LES design analyses documentation shall include:

- Definition of the objective of the analyses,
- Definition of design inputs and their sources,
- Results of literature searches or other applicable background data,
- Identification of assumptions and designation of those that must be verified as the design proceeds,
- Identification of any computer calculation, including computer type, computer program (e.g., name), revision identification, inputs, outputs, evidence of reference to computer program verification and the bases (or reference thereto) supporting application of the computer program to the specific physical problem,
- Review and approval.

DESIGN VERIFICATION

The following design control requirements shall be applied to verify the adequacy of LES design:

- LES design verification is required for design documents, and shall be performed using one or a combination of the design review, alternate calculations and/or qualification testing methods.
- The particular design verification method used shall be documented.
- Results of design verification shall be documented and shall include the identification of the verifier(s).
- Competent individuals or groups, other than those, who performed the original design (but may be from the same organization), shall perform design verification. If necessary, this verification may be performed by the originator's supervisor provided that the engineering supervisor did not specify a singular design approach or rule out certain design considerations and did not establish the design inputs used in the design; or the supervisor is the only individual in the organization competent to perform the verification.

LES design verification shall be performed in a timely manner at appropriate times during the design process. Verification shall be performed before release for procurement, manufacture or construction, or release to another organization for use in other design work. In some cases (such as when insufficient data exists) it may be necessary to release unverified designs to other engineering organizations or disciplines to support schedule requirements. Unverified portions of the design shall be clearly identified and procedurally controlled. In all cases, design verification shall be completed before relying on the item or computer program to perform its function. The extent of design verification required shall be a function of the importance to safety, complexity of design, degree of standardization, state of the art and similarity with previously proven designs.

LES use of previously standardized designs shall be controlled according to the following requirements:

- The applicability of standardized or previously proven designs shall be verified with respect to meeting pertinent design inputs for each application.
- Known problems affecting standard or previously proven designs and their effects on other features shall be considered.
- The “Americanization” of previously proven European designs shall be documented in accordance with the applicable QA procedure.
- The original design and associated verification measures shall be adequately documented and referenced in the files for subsequent application of the design.
- Changes in previously verified designs shall require re-verification. Such verifications shall include the evaluation of the effects of those changes on the overall previously verified design and on any design analyses upon which the design is based.

DESIGN VERIFICATION METHODS

Acceptable verification methods include, but are not limited to, any one of the following or a combination of the following:

- Design Reviews
- Alternate Calculations
- Qualification Testing

DESIGN REVIEWS

Design reviews are critical reviews to provide assurance that the final design is correct and satisfactory. The following items shall be addressed, as applicable during the review:

- Were the design inputs correctly selected and incorporated into the design?
- Are assumptions necessary to perform the design activity adequately described, reasonable and, where necessary, re-verified?
- Was an appropriate design method used?
- Is the design output reasonable compared to the applicable design inputs?
- Are the necessary design input and verification requirements for interfacing organizations specified in the design documents or in supporting procedures and instructions?

ALTERNATE CALCULATIONS

The appropriateness of assumptions, input data, and the computer program or other calculation methods used, shall be evaluated and the results shall be checked through the use of alternate calculation methods to verify the correctness of the original calculations or analyses.

QUALIFICATION TESTS

If design adequacy is to be verified by qualification testing, the tests shall be identified, procedurally controlled and documented according to the following:

- The test configuration shall be defined and documented.
- Testing shall demonstrate the adequacy of performance under conditions that simulate the most adverse design conditions. Operating modes and environmental conditions in which the item must perform satisfactorily shall be considered in determining the most adverse design conditions.
- If the tests verify only specific design features, then the other features of the design shall be verified by other means.
- Test results shall be documented and evaluated to ensure that test requirements have been met.
- If qualification testing indicates that a modification to an item is necessary to obtain acceptable performance, then the modification shall be documented and the item modified and re-tested or otherwise verified to ensure satisfactory performance.
- Scaling laws shall be established, verified and documented when tests are being performed on models or mockups.
- The results of model test work shall be subject to error analysis, where applicable, before using the results in final design work.

DESIGN CHANGE CONTROL

Design changes during the initial design phase and the operational phase shall be controlled according to the following requirements:

- Changes to final designs, field changes, modifications to the operating facility and nonconforming items dispositioned as "use-as-is" or "repair," as described in Section 15, Nonconforming Items, and shall have documented justification for use and are subject to the same design control measures and reviews as those applied to the original design.
- Design control measures for changes shall include provisions to ensure that the design analyses for the item are still valid.
- Changes shall be reviewed and approved by the affected groups or organizations that reviewed and approved the original design documents, with the following clarifications:
    - If the organization that originally was responsible for approving a particular design document is no longer responsible, then a new responsible organization shall be designated.
    - The designated organization shall have demonstrated competence in the specific design area of interest and have an adequate understanding of the requirements and intent of the original design.

- The interface between the design organization responsible for finalizing a design change and other organizations either involved in the review of the change, such as the QA and configuration management organizations, and those affected by the change, such as the operations and maintenance organizations, described in the next subsection, Design Interface Control, shall be maintained.
- The design process and design verification practices and procedures shall be reviewed and modified, as necessary, when a significant design change is required because of an incorrect design. These design deficiencies shall be documented according to Section 16.0, Corrective Actions. If these deficiencies cause constructed or partially constructed items (systems, structures or components) to be deficient, the affected items shall be controlled in accordance with Section 15, Nonconforming Items.
- When a design change is approved other than revision to the affected design documents, field changes shall be incorporated into affected design documents when such incorporation is appropriate.

DESIGN INTERFACE CONTROL

LES design interfaces shall be identified and procedurally controlled. Design efforts shall be coordinated among interfacing organizations as detailed in LES procedures. Interface controls shall include the assignment of responsibility and the establishment of procedures among interfacing design organizations for the review, approval, release, distribution and revision of documents involving design interfaces. LES design information transmitted across interfaces shall be documented and procedurally controlled. LES transmittals of design information and/or documents shall reflect the status of the transmitted information and documents. Incomplete designs that require further evaluation, review or approval shall be identified. When it is necessary to initially transmit the design information orally or by other informal means, design information shall be promptly confirmed through a controlled document.

During the operational phase, the Plant Manager is responsible for ensuring the facility complies with all applicable regulatory requirements including the requirements of this QA Program. In the discharge of these responsibilities, the Plant Manager directs the activities of the Technical Services, which includes Engineering and Maintenance, and Operations. Procedures for controlling the interfaces and configuration management ensure that changes and modifications are properly managed and disseminated to those responsible personnel or organizations whose duties may be affected by the design change or modification and do not adversely impact the safe operation of the plant.

COMPUTER SOFTWARE CONTROLS

If LES uses software to produce or manipulate data that is used directly in the design, analysis and operation of structures, systems, and components relied on for safety, the provisions provided in Part II ASME NQA-1-1994 Subpart Part 2.7, *Quality Assurance Requirements of Computer Software for Nuclear Facility Applications,* as revised by NQA-1a-1995 Addenda of NQA-1-1994 and ASME NQA-1-1994, Part I, Supplement 11S-2, *Supplementary Requirements for Computer Program Testing* shall apply. Procedures will be developed to implement of these provisions as applicable.

DOCUMENTATION AND RECORDS

Design documentation which provide evidence that the design and design verification were performed in accordance with this QAPD shall be collected and maintained in accordance with the requirements of Section 17 Quality Assurance Records. The documentation shall include not only final design documents such as drawings, specifications and revision thereto but also documentation which identifies the important steps, including sources of design inputs that support the final design.

SECTION 4 PROCUREMENT DOCUMENT CONTROL

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 4, Procurement Document Control, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 4 and Supplement 4S-1 of NQA-1-1994.

LES procurements shall be issued only to those suppliers that have been evaluated and qualified as acceptable for the particular scope of material, equipment and services to be procured. The material, equipment and services shall be procured from approved suppliers by procurement documents, approved by the LES President and QA Director or their qualified designees. Applicable design bases and other requirements necessary to assure adequate quality shall be included or referenced in documents for procurement of items and services. Procurement documents shall require suppliers to have a quality assurance program consistent with the applicable requirements of 10 CFR 50 Appendix B and this QAPD. The requirements of 10 CFR 21, Reporting of Defects and Noncompliance, are invoked during design, construction, testing and operations of QA Level 1 procurement or dedication of items and services including the dedication of items or services used to satisfy the requirements of 10 CFR 50, Appendix B or 10 CFR 70, Domestic Licensing of Special Nuclear Material. LES will also apply the requirements of 10 CFR 21 where appropriate, regardless of QA level.

Procurement Document Content

LES procurement documents issued for QA Level 1 items or services shall include the following provisions, as applicable to the procured material, equipment or service:

- Statement of the scope of work to be performed by the supplier.
- Technical requirements including:
  - Design bases, identified or referenced in the procurement documents.
  - Specific documents (such as drawings, codes, standards, regulations, procedures or instructions) describing the technical requirements of the material, equipment or services to be furnished, shall be specified along with their revision level or change status.
  - Tests, inspections or acceptance requirements that LES will use to monitor and evaluate the performance of the supplier shall be specified.
- Quality Assurance Program requirements including:
  - A requirement for the supplier to have a documented quality assurance program that implements applicable requirements of 10 CFR 50, Appendix B and this QAPD in place before the initiation of work. The extent of the quality assurance program shall depend on the scope, nature or complexity of the material, equipment or service to be procured. The supplier shall also incorporate the appropriate requirements into any subtier supplier issued procurement documents.
  - A requirement invoking NRC reporting requirements of 10 CFR 21 for QA Level 1 procurements.
- Right of access to supplier, including subtier, facilities and records for inspection or audit by LES, or other designee authorized by LES.
- Provisions for establishing witness/inspection hold points beyond which work cannot

proceed by the supplier without LES QA Director authorization. The LES Engineering and Contracts Manager may also establish hold points indicating work that cannot proceed without authorization by the Engineering and Contracts Manager.

- Documentation required to be submitted to LES for information, review or acceptance shall be identified along with a document submittal schedule. Record retention times, disposition requirements and record maintenance responsibilities shall be identified for documentation that will become quality assurance records.
- Requirements for the supplier to report to LES in writing adverse quality conditions resulting in work stoppages and nonconformances. LES approval of partial and full work releases and disposition of nonconformances is required.
- Identification of any spare and replacement parts or assemblies and the appropriate delineation of technical and quality assurance data required for ordering these parts or assemblies. Commercial Grade procurements shall also be identified in procurement documents.

Procurement Document Review and Approval

Procurement document reviews shall be performed and documented before issuing the procurement documents to the supplier. A review of the procurement documents and any changes thereto shall be made to verify that documents include all applicable requirements specified under Section 4, Procurement Document Content, above and contain appropriate provisions to ensure that material, equipment or services will meet the governing requirements. Reviews shall be performed and documented to provide objective evidence of satisfactory accomplishment of such review prior to contract award. Changes made as a result of the bid evaluation or precontract negotiations shall be incorporated into the procurement documents. The review of such changes and their effects shall be completed prior to contract award. This review shall include the following considerations: 1) appropriate requirements specified in Procurement Document Content above, 2) a determination of any additional or modified design criteria, and 3) an analysis of exceptions or changes requested by the supplier and a determination on the impacts such changes may have on the intent of the procurement documents or quality of the item or service to be provided shall be performed by the LES organization initiating the procurement. Personnel who have access to pertinent information and have an adequate understanding of the requirements and scope of the procurement shall perform reviews of the procurement documents. Reviewers shall include representatives from the Engineering and Contracts and QA organizations. The QA review shall assure compliance to quality assurance requirements.

Procurement Document Change

Changes to the scope of work, technical requirements, quality assurance program requirements, right of access, documentation requirements, work stoppage and nonconformance, hold points and lists of spare and replacement parts delineated in procurement documents, shall be subject to the same degree of control as used in the preparation of the original procurement document.

SECTION 5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 5, Instructions, Procedures, and Drawings, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 5 of NQA-1-1994 Part I.

Activities affecting quality shall be prescribed by and conducted in accordance with approved procedures and other implementing documents (drawings, specifications, etc.) appropriate to the circumstances. Generally, four types of procedures are used by LES to ensure that activities are carried out in compliance with the requirements of this QAPD and in a safe manner. These include administrative, operating, maintenance and emergency procedures. Administrative procedures would include areas such as engineering procurement, etc. Administrative procedures are the higher level procedures that prescribe the implementation of the requirements provided in this QAPD. Operating and maintenance procedures are utilized to implement the QA program during the start up, operation, and testing of the facility. During the design and construction phases, procedures are reviewed and approved by the affected organizations with review and oversight by the QA organization. Those procedures that delineate the responsibilities and functions of the QA organization, the QA procedures, are approved by the LES QA Director to ensure compliance with QAPD. During operations, the LES QA Manager and Plant Manager have responsibility to review and approve the procedures that cover activities under their organizational purview that relate to the QAPD and the safe operation of the plant. Procedures approved by the Plant Manager will be subject to selected review and oversight by the QA organization.

TYPES OF DOCUMENTS

The type of document to be used to perform work shall be appropriate to the nature and circumstances of the work being performed. Documents include procedures, drawings and specifications. Work controlling procedures may also utilize approved checklists, travelers or other means to assure process requirements are met including prerequisite requirements prior to starting work. Procedures provide a consistent method for process performance and documentation of completion as well as ensure specified safety and environmental conditions are maintained.

CONTENT OF DOCUMENTS

Documents shall include or reference the following information as appropriate to the work to be performed:

- Responsibilities of the organizations affected by the document,
- Quality, technical and regulatory requirements,
- A sequential description of the work to be performed including controls for altering the sequence of required inspections, tests and other operations,
- Quantitative or qualitative acceptance criteria sufficient for determining that prescribed activities have been satisfactorily accomplished,
- Prerequisites, limits, precautions, process parameters and environmental conditions,
- Quality verification points and hold points,

- Methods for demonstrating that the work was performed as required,
- Identification of the lifetime or nonpermanent quality assurance records generated by the implementing document, and
- Identification of associated QA Levels as appropriate.

REVIEW, APPROVAL, AND CONTROL OF DOCUMENTS

Procedures and implementing documents shall be controlled according to the requirements of Section 6, Document Control of this document. Procedures and implementing documents shall be reviewed and approved as described in this section and in Section 6.

SECTION 6 DOCUMENT CONTROL

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 6, Document Control, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 6 and Supplement 6S-1 of NQA-1-1994.

Procedures are established which control the preparation, issuance and changes of documents that specify quality requirements or prescribe activities affecting quality. Measures are established to ensure that documents, including revisions are adequately reviewed, approved, and released for use by authorized personnel. Controlled documents are transmitted to the appropriate locations where the prescribed activity is being performed. Superceded documents are destroyed or retained only when they have been properly marked.

TYPES OF DOCUMENTS

QA procedures, other administrative procedures and implementing documents and documents specifying quality requirements or prescribing activities affecting quality shall be controlled in accordance with this section. LES documents controlled under the LES QA Program will be specified by procedures and include, but are not limited to, procedures, design requirements document, design basis documents, engineering specifications, instructions, drawings, calculations, procurement documents, and documents that need to be controlled due to being input to other LES design documents or used for construction and operations affecting quality.

PREPARING AND REVIEWING DOCUMENTS

The document control system shall ensure that the identification of documents to be controlled and their specified distribution are preceduralized. The system shall further ensure that the responsibility for preparing, reviewing, approving and issuing documents shall be assigned by procedure to the appropriate LES functional area manager. Implementing documents and documents specifying quality requirements or prescribing activities affecting quality, shall be reviewed in accordance with applicable procedures for adequacy, correctness and completeness and by the QA organization as specified by procedure, prior to approval and issuance. The organizational position(s) responsible for approving the document(s) for release shall be identified in the applicable procedures.

CONTROLLING THE DISTRIBUTION AND USE OF DOCUMENTS

Documents needing to be placed under the document control system are transmitted to the Document Control organization with the distribution list for document holders. The Document Control organization shall enter the document into the Document Control electronic database and master list of controlled documents, assign document control numbers, complete transmittal forms and distribute the documents and transmittal form to the document holders. Document holders shall acknowledge receipt on the transmittal and send the acknowledgement to the Document Control organization. The up-to-date master listing of controlled documents will be made continuously available to document holders to verify that they have the current revisions. The document control process will be audited in accordance with the requirements of Section 18, QA Audits, to verify implementation effectiveness.

CHANGES TO DOCUMENTS

Changes to documents other than minor changes shall be reviewed for adequacy, correctness and completeness, prior to approval and issuance. Major changes shall be reviewed and approved by the same organization that performed the original review and approval unless other organizations are specifically designated. The reviewing organization shall have access to the applicable background data or information upon which to base their approval. A temporary procedure change that does not change the intent of the procedure may be made at the work location by responsible management. The applicable procedure shall control the process, documentation and approval of the temporary changes.

MINOR CHANGES

Minor changes such as inconsequential editorial corrections may be made to documents without being subject to the review and approval of the requirements specified above. The applicable procedure shall define the organizational positions authorized and criteria acceptable for making minor changes.

SECTION 7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT AND SERVICES

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 7, Control of Purchased Material, Equipment and Services, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 7 and Supplement 7S-1 of NQA-1-1994 Part I as revised by NQA-1a-1995 Addenda of NQA-1-1994.

LES procurement of material, equipment and services is controlled to assure conformance with specified requirements. These controls include requirements for pre-award evaluations of suppliers' QA programs, annual evaluations, periodic audits/source inspections and surveillance. Suppliers with a LES approved QA program are placed on the LES ASL prior to award of contract. Source inspections and surveillances, evaluation of objective evidence of quality furnished by the supplier, maintaining the ASL, as well as, examination of received items and services are the responsibility of LES QA organization and are performed, as necessary, upon delivery or completion to ensure requirements specified in procurement documents are met. Supplier evaluations, annual evaluations, audits, surveillances, source inspections and receipt inspections shall be documented.

PROCUREMENT PLANNING

LES procurements shall be planned and documented to ensure a systematic approach to the procurement process exists and supports the schedule. Procurement planning shall:

- Identify procurement methods and organizational responsibilities, including what is to be accomplished, who is to accomplish it, how it is to be accomplished, and when it is to be accomplished.
- Identify and document the sequence of actions and milestones needed to effectively complete the procurement.
- Provide for the integration of the following activities:
    - Procurement document preparation, review and change control according to the requirements of Section 4, Procurement Document Control
    - Selection of procurement sources, proposal/bid evaluation and award
    - LES evaluation of supplier performance
    - LES verifications including any hold and witness point notifications
    - Control of nonconformances
    - Corrective action
    - Acceptance of the material, equipment or service
    - Identification of quality assurance records to be provided to LES.
- Be accomplished as early as possible, and no later than at the start of those procurement activities that are required to be controlled to assure interface compatibility and a uniform approach to the procurement process.
- Be performed relative to the level of importance, complexity and quantity of the item or service being procured and the supplier's quality performance.

- Include the involvement of the LES QA organization to ensure that the QA requirements have been properly identified.

SOURCE EVALUATION AND SELECTION

Supplier selection shall be based on an evaluation, performed before the contract and/or purchase order is awarded, of the supplier's capability to provide items or services in accordance with procurement document (technical and quality) requirements. The functional area needing the procurement shall request that the LES QA organization evaluate the potential supplier for placement on the LES ASL. Responsibilities and measures for evaluating and selecting procurement sources are detailed in the applicable QA procedure and include one or more of the following methods for evaluating potential suppliers:

- Evaluation of the supplier's history for providing an identical or similar product that performs satisfactorily in actual use. The supplier's history shall reflect current capability.
- Evaluation of supplier's current quality assurance records supported by any documented qualitative and quantitative information which can be objectively evaluated.
- Evaluation of the supplier's technical and quality capability based on an evaluation of supplier facilities, personnel and quality assurance program implementation.

The results of procurement source evaluation and selection shall be documented in accordance with the applicable QA procedure.

PROPOSAL/BID EVALUATION

For proposals and bids, technically qualified personnel from the QA and Engineering and Contracts or other affected/involved organizations shall perform an evaluation to determine if the proposal/bid meets procurement document requirements. As a minimum, this evaluation shall review the following subjects consistent with the importance, complexity and quantity of items or services being procured:

- Technical considerations
- QA program requirements
- Supplier personnel qualifications
- Supplier production capability and past performance
- Alternatives and exceptions

Before the contract is awarded, the LES QA Director or Engineering and Contracts Manager, or other affected/involved organization manager shall resolve, or obtain commitments to resolve, unacceptable quality conditions identified during the proposal/bid evaluation. Supplier quality assurance programs shall be evaluated by the QA organization before contract placement, and any deficiencies that would affect quality shall be corrected before starting work subject to these requirements. Supplier QA programs shall be accepted by the LES QA Director before the supplier starts work.

SUPPLIER PERFORMANCE EVALUATION

The LES Engineering and Contracts Manager in coordination with the QA Director shall establish measures to routinely interface with the supplier and to verify supplier performance. The measures shall include:

- Establishing an understanding between LES and the supplier of the requirements and specifications identified in procurement documents.
- Requiring the supplier to identify planning techniques and processes to be used in fulfilling procurement document requirements.
- Reviewing supplier documents that are prepared or processed during work performed to fulfill procurement requirements.
- Identifying and processing necessary change information.
- Establishing the method to be used to document information exchanges between LES and supplier.
- Establishing the extent of source surveillance and inspection.

The extent of LES verifications shall be a function of the relative importance, complexity/quantity of items or services being procured and the supplier's quality performance. Verification activities shall be accomplished by qualified personnel assigned to check, inspect, audit, or witness the activities of the suppliers. LES verifications shall be conducted as early as practical and shall not relieve the supplier of the responsibility for the verification of quality achievement. Verifications shall include supplier audits, surveillances or source inspections (or combinations) used as a method of evaluating the supplier's performance, and evaluation of purchaser's documentation to aid in the determination of the effectiveness of the supplier's quality assurance program. Records, including source surveillances and inspections, audits, receiving inspections, nonconformances, dispositions, waivers, and corrective actions shall be maintained in accordance with the requirements of Section 17, Quality Assurance Records.

CONTROL OF SUPPLIER GENERATED DOCUMENTS

Supplier generated documents shall be controlled, processed and accepted by LES in accordance with the requirements established in the applicable QA procedures. Measures shall be implemented to ensure that the submittal of supplier generated documents is accomplished in accordance with the procurement document requirements. These measures shall also provide for the acquisition, processing and recorded evaluation of technical, inspection and test data compared against the acceptance criteria.

CONTROL OF CHANGES IN ITEMS OR SERVICES

LES shall establish contractual controls with suppliers to ensure that changes in procurement documents are controlled and documented in accordance with this QAPD.

ACCEPTANCE OF ITEMS OR SERVICES

Methods for accepting supplier furnished material, equipment or services shall include one or more of the following, as appropriate to the items or services being procured:

- Evaluating the supplier certificate of conformance,
- Performing one or a combination of source verification, receiving inspection or post-installation test,

- Technical verification of the data produced (services only),
- Surveillance or audit of the activities (services only),
- Review of objective evidence for conformance to procurement requirements (services only).

The supplier shall verify that furnished material, equipment or services comply with LES's procurement requirements before offering the material, equipment or services for acceptance and shall provide to LES objective evidence that material, equipment or services conform to procurement documents. Where required by code, regulations or contract provisions, documentary evidence that items conform to procurement documents shall be available at the site prior to installation or use.

CERTIFICATE OF CONFORMANCE

When a certificate of conformance is used to accept material, equipment or service:

- The certificate shall identify the purchased material, equipment or service to the specific procurement document.
- The certificate shall identify the specific procurement requirements met by the purchased material, equipment or service. The procurement requirements identified shall include any approved changes, waivers or deviations applicable to the material, equipment or service.
- The certificate shall identify any procurement requirements that have not been met together with an explanation and the means for resolving nonconformances.
- The certificate shall be signed and dated or otherwise authenticated by an individual who is responsible for the supplier's quality assurance function and whose responsibilities and position are described in the supplier's quality assurance program.
- The certification process, including the implementing documents to be followed in filling out a certificate and the administrative implementing documents for review and approval of the certificates, shall be described in the supplier's quality assurance program.
- Measures shall be identified to verify the validity of supplier certificates and the effectiveness of the certification process (such as by audit of the supplier or by an independent inspection or test of the item). Verifications shall be conducted by LES at intervals commensurate with the past quality performance of the supplier.

SOURCE VERIFICATION

LES may accept material, equipment or service by monitoring, witnessing or observing activities performed by the supplier. This method of acceptance is called source verification. Source verification shall be implemented consistent with the supplier's planned inspections, examinations or tests at predetermined points and performed at intervals consistent with the importance and complexity of the item. Documented evidence of acceptance of source verified material, equipment or services shall be furnished to the receiving destination of the item, to LES, and to the supplier. Personnel qualified in accordance with the applicable requirements for the material, equipment or service being procured shall perform source verification.

RECEIVING INSPECTION

When receiving inspection is used to accept an item:

- The inspection shall consider any source verifications/audits and the demonstrated quality

performance of the supplier.

- The inspection shall be performed in accordance with established inspection procedures.
- The inspection shall verify, as applicable, proper configuration; identification; dimensional, physical and other characteristics; freedom from shipping damage; and cleanliness.
- The inspection shall be planned and executed according to the requirements of Section 10 Inspection.
- Receiving inspection shall be coordinated with a review for adequacy and completeness of any required supplier documentation submittals.

POST-INSTALLATION TESTING

When post-installation testing is used as a method of acceptance, the LES Engineering and Contracts Manager or the affected/involved LES organization manager and the supplier, when possible, shall mutually establish test requirements and acceptance documentation. The LES Engineering Contracts Manager is ultimately responsible for ensuring appropriate test requirements and acceptance documentation are established.

CONTROL OF SUPPLIER NONCONFORMANCES

The LES Engineering and Contracts organization and the supplier shall establish and document the process for disposition of items that do not meet procurement document requirements. The supplier shall evaluate nonconforming items according to the applicable requirements of Section 15, Nonconforming Items and submit a report of nonconformance to LES Engineering and Contracts organization including supplier recommended disposition (for example, use-as-is or repair) and technical justification. Reports of nonconformances to procurement document requirements, or documents approved by LES, shall be submitted to LES Engineering and Contracts organization for approval of the recommended disposition whenever one of the following conditions exists:

- Technical or material requirements are violated.
- A requirement in supplier documents, which have been approved by LES, is violated.
- The nonconformance cannot be corrected by continuation of the original manufacturing process or by re-work.
- The item does not conform to the original requirement even though the item can be restored to a condition such that the capability of the item to function is unimpaired.

LES Engineering and Contracts organization shall disposition the supplier's recommendation and verify implementation of the disposition. LES will maintain records of the supplier-submitted nonconformances.

COMMERCIAL GRADE ITEMS

Where the design utilizes commercial grade material and/or equipment, the following requirements are an acceptable alternate to other requirements of this Section:

- The commercial grade material/equipment is identified in an approved design output document. An alternate commercial grade material/equipment may be applied, provided there is verification that the alternate commercial grade material/equipment will perform the intended function and will meet design requirements applicable to both the replaced

material/equipment and its application.

- Supplier evaluation and selection, where determined necessary by the LES based on complexity and importance to safety, shall be in accordance with *Source Evaluation and Selection* section of this document.
- Commercial grade items shall be identified in the purchase order by the manufacturer's published product description (e.g., catalog number).
- One or a combination of the following methods shall be utilized to provide reasonable assurance that the item meets the acceptance criteria for the characteristics identified to be verified for acceptance:
  - special test(s) or inspection (s) or both;
  - commercial grade survey of the supplier;
  - source verification;
  - acceptable supplier/item performance records.
- Prior to acceptance of a commercial grade item, LES QA organization shall determine that:
  - damage was not sustained during shipment;
  - the item received has satisfied the specified acceptance criteria;
  - inspection and/or testing is accomplished, as required, to assure conformance with critical characteristics; and
  - documentation, as applicable to the item, was received and is acceptable.

APPROVED SUPPLIER LIST

The LES Quality Assurance Director is responsible for the development and maintenance of the LES ASL. The ASL contains those suppliers with acceptable QA Programs that have been evaluated and accepted by the LES QA in accordance with approved procedures. The LES QA organization shall perform and document an evaluation of each supplier every 12 months. Satisfactory results will allow the supplier to remain on the ASL. Additionally, suppliers will be evaluated by means of an audit at least triennially, if initial approval was by audit or survey. Suppliers that have unacceptable evaluations or that have not had a procurement placed with them in three years will be removed from the ASL.

SECTION 8 IDENTIFICATION AND CONTROL MATERIALS, PARTS AND COMPONENTS

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 8, Identification and Control of Materials, Parts and Components, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 8 and Supplement 8S-1 of NQA-1-1994 Part I as revised by NQA-1a-1995 Addenda.

The controls necessary to ensure that only correct and accepted items are used or installed will be required by the appropriate QA procedure. Identification requirements for materials, parts and components are stated in design specifications, drawings, and procurement documents. Specific identification requirements are as follows.

- Identification markings, when used shall be applied using materials and methods which provide a clear and legible identification and do not detrimentally affect the function or service life of the item. Markings shall be transferred to each part of an item when subdivided and shall not be obliterated or hidden by surface treatments or coatings unless other means of identification are substituted.
- When required by specifications or codes and standards, identification of material or equipment with traceability to the corresponding mill test reports, certifications and other required documentation is maintained throughout fabrication, erection, installation, or use.
- Sufficient precautions shall be taken to preclude identifying materials in a manner that degrades the function or quality of the item being identified.

Control of material, parts and components is governed by approved procedures. Specific control requirements include the following.

- Identification of nonconforming or rejected materials, parts or components to ensure that they are not inadvertently used.
- Verification of correct identification of materials (including consumable materials or items with a limited shelf life), parts, and components shall be required to prevent the use of incorrect or defective items.
- Receipt inspection to ensure that materials, parts or components are properly identified and that supporting documentation is available as required by procurement specifications.
- Maintaining and replacement of markings and identification records due to damage during handling, aging or environmental exposure.

SECTION 9 CONTROL OF SPECIAL PROCESSES

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 9, Control of Special Processes, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 9 and Supplement 9S-1 of NQA-1994 Part I.

Processes affecting the quality of items or services shall be controlled by written procedures using drawings, checklists, travelers or other appropriate means. These means shall ensure that the process parameters are controlled and that specified environmental conditions are maintained. Special processes that control or verify quality, such as those used in welding, heat treating, and nondestructive examination, shall be performed by qualified personnel using qualified procedures in accordance with specified requirements.

SPECIAL PROCESSES

Special processes that control or verify quality shall be controlled according to the requirements of this section whether or not they are covered by existing codes and standards, or whether or not the quality requirements specified for an item exceed those of existing codes or standards.

PERSONNEL, IMPLEMENTING DOCUMENTS, AND EQUIPMENT QUALIFICATIONS

Implementing LES documents shall be used to ensure that process parameters are controlled and that the specified environmental conditions are maintained. Each special process shall be performed in accordance with appropriate implementing documents and these implementing documents shall include or reference:

- The responsibility of the organization performing the special process to adhere to the approved procedures and processes,
- Qualification requirements for personnel, implementing documents and equipment,
- Conditions necessary for accomplishment of the special process. These conditions shall include proper equipment, controlled parameters of the process and calibration requirements, and/or
- Requirements of applicable codes and standards, including acceptance criteria for the special process.

QUALIFICATION OF NONDESTRUCTIVE EXAMINATION PERSONNEL

Personnel who have been qualified and certified in accordance with Section 2.0, QA Program, of this QAPD shall perform nondestructive examinations required for the LES work activities.

DOCUMENTATION

Records shall be maintained as appropriate in accordance with Section 17, Quality Assurance Records, for currently qualified personnel, processes and equipment of each special process.

SECTION 10 INSPECTION

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 10, Inspection, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 10 and Supplement 10S-1 of NQA-1-1994 Part I.

Inspections required to verify conformance of an item or activity to specified requirements are planned and executed. Characteristics to be inspected and inspection methods to be employed are specified in procedures. Inspection results are documented. Persons other than those who performed or directly supervised the work being inspected shall perform inspection for acceptance. Inspection requirements and acceptance criteria shall include specified requirements contained in the applicable design documents or other pertinent technical documents approved by the responsible design organization. Inspection activities are documented and controlled by instructions, procedures, drawings, checklists, travelers or other appropriate means.

INSPECTION PLANNING

Inspection planning shall be performed, documented and include:

- Identification of each work operation where inspection is necessary to ensure quality and implementing documents that shall be used to perform the inspections;
- Identification of the characteristics to be inspected and the identification of when, during the work process, inspections are to be performed;
- Identification of inspection or process monitoring methods to be employed;
- The final inspection shall be planned to arrive at a conclusion regarding conformance of the item to specified requirements;
- Identification of the functional qualification level (category or class) of personnel performing inspections;
- Identification of acceptance criteria;
- Methods to record objective evidence of inspection results; and
- Selection and identification of the measuring and test equipment to be used to perform the inspection.

SELECTING INSPECTION PERSONNEL TO PERFORM INSPECTION

The individual who performs an inspection to verify conformance of an item to specified acceptance criteria shall be qualified to perform the assigned inspection tasks in accordance with the requirements of Section 2, QA Program. Data recorders, equipment operators or other inspection team members who are supervised by a qualified inspector shall not be required to be a qualified inspector. Verification of conformance shall be by a qualified person. Inspections shall be performed by personnel other than those who performed or directly supervised the work being inspected. Inspection personnel shall not report directly to the immediate supervisors who are responsible for performing the work being inspected.

INSPECTION HOLD POINTS

When mandatory hold points are used to control work that shall not proceed without the specific

consent of the organization placing the hold point, the specific hold points shall be indicated in implementing documents. Consent to waive specified hold points shall be documented and approved before continuing work beyond the designated hold point.

STATISTICAL SAMPLING

When statistical sampling is used to verify the acceptability of a group of items, the statistical sampling method used shall be based on recognized standard practices and these practices shall be implemented through applicable approved procedures.

IN-PROCESS INSPECTIONS AND MONITORING

Items shall be inspected when necessary to verify quality. If inspection of processed items is impossible or disadvantageous, indirect control by monitoring of processing methods, equipment and personnel shall be provided. Inspection and process monitoring shall be conducted when control is inadequate with only one method. A combination of inspection and process monitoring methods, when used, shall be performed in a systematic manner to ensure that the specified requirements for control of the process and the quality of the item are met throughout the duration of the process. Controls shall be established and documented for the coordination and sequencing of inspections and monitoring at established inspection points during successive stages of the process or construction.

FINAL INSPECTION

Finished items shall be inspected for completeness, markings, calibration, adjustments, protection from damage or other characteristics as required in order to verify the quality and conformance of the item to specified requirements. Documentation not previously examined shall be examined for adequacy and completeness. The final inspection shall be planned to arrive at a conclusion regarding conformance of the item to specified requirements. Final inspections shall include a review of the results and resolution of any nonconformances identified by earlier inspections. Modifications, repairs or replacements of items performed subsequent to final inspection shall require re-inspection or retest, as appropriate, to verify acceptability.

ACCEPTING ITEMS

The acceptance of an item shall be documented and approved by qualified and authorized personnel. The inspection status of an item shall be identified according to Section 14, Inspection, Test and Operating Status.

INSERVICE INSPECTION

Inservice inspection or surveillance of structures, systems, or components shall be planned and implemented by or for the LES Operating organization. Procedures shall control the inspections to verify that the characteristics of the item remain with the specified limits. The inspection procedure shall include the following, as appropriate:

- Evaluations of performance capabilities of essential emergency and safety systems and equipment,
- Verification of calibration and integrity of instruments and instrument systems, and
- Verification of maintenance.

INSPECTION DOCUMENTATION

Inspection documentation shall identify:

- The item inspected, date of inspection, the name of the inspector who documented, evaluated and determined acceptability;
- Name of data recorder, as applicable and type of observation or method of inspection;
- The inspection criteria, sampling plan or reference documents (including revision levels) used to determine acceptance;
- Results or acceptability of characteristics inspected;
- Measuring and test equipment used during the inspection including the identification number and the most recent calibration date; and
- Reference to information on actions taken in connection with nonconformances, as applicable.

SECTION 11 TEST CONTROL

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 11, Test Control, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 11 and Supplement 11S-1 of NQA-1-1994 Part I. The commitment to the provisions in Supplement 11S-2, Supplementary Requirements for Computer Program Testing is addressed in Section 3, Design Control.

Tests required to verify conformance of an item or computer program to specified requirements and to demonstrate satisfactory performance for service are planned and executed. Characteristics to be tested and test methods to be employed are specified. Test results are documented and their conformance with acceptance criteria is evaluated. Tests required to collect data, such as for siting or design input, shall be planned, executed, documented and evaluated.

TEST REQUIREMENTS

Test requirements and acceptance criteria shall be provided or approved by the organization responsible for the design of the item to be tested unless otherwise designated. Required tests, including, as appropriate, prototype qualification tests, production tests, proof tests prior to installation, construction tests, pre-operational tests, and operational tests are controlled. Test requirements and acceptance criteria are based upon specified requirements contained in applicable design or other pertinent technical documents.

TEST PROCEDURES

Test procedures shall include:

- Test objectives and the identification of any implementing documents to be developed to control and perform tests as appropriate;
- Identification of items to be tested, test requirements and acceptance limits, including required levels of precision and accuracy;
- Identification of test methods to be employed and instructions for performing the test;
- Test prerequisites that address calibrated instrumentation, appropriate and adequate test equipment/instrumentation, trained personnel, condition of test equipment and the item to be tested, suitably controlled environmental conditions and provisions for data acquisition;
- Mandatory hold points and methods to record data and results;
- Provisions for ensuring that prerequisites for the given test have been met;
- Selection and identification of the measuring and test equipment to be used to perform the test to ensure that the equipment is of the proper type, range, accuracy, and tolerance to accomplish the intended function; and
- Identification of the functional qualification level of personnel performing tests.

PERFORMING TESTS

Tests shall be performed in accordance with procedures that address the following requirements as applicable:

- Provisions for determining when a test is required, describing how tests are performed, and ensuring that testing is conducted by trained and appropriately qualified personnel.
- Include or reference test objectives and provisions for ensuring that prerequisites for the given test have been met, adequate calibrated instrumentation is available and used, necessary monitoring is performed and suitable environmental conditions are maintained.
- Test requirements and acceptance criteria provided or approved by the organization responsible for the design of the item to be tested, unless otherwise designated.
- Test requirements and acceptance criteria based upon specified requirements contained in applicable design or other pertinent technical documents.
- Potential sources of uncertainty and error. Test parameters affected by potential sources of uncertainty and error shall be identified and controlled.

MONITORING AND OVERSIGHT OF SUPPLIER TEST

The LES Engineering and Contracts Manager in coordination with the QA Director shall establish measures to routinely interface with the supplier and to verify supplier performance. LES may accept material, equipment or service by monitoring, witnessing or observing activities performed by the supplier. This method of acceptance is called source verification. Source verification shall be implemented consistent with the supplier's planned inspections, examinations or tests at predetermined points and performed at intervals consistent with the importance and complexity of the item. Documented evidence of acceptance of source verified material, equipment or services shall be furnished to the receiving destination of the item, to LES, and to the supplier. Personnel qualified in accordance with the applicable requirements for the material, equipment or service being procured shall perform source verification.

USE OF OTHER TESTING DOCUMENTS

Other testing documents (e.g., American Society for Testing and Materials (ASTM)) specifications, supplier manuals or other related documents containing acceptance criteria may be used instead of preparing special test procedures. If used, the information shall be incorporated by reference in the approved test procedure. Implementing documents shall include adequate supplemental instructions as required to ensure the required quality of the testing work.

TEST RESULTS

Test results shall be documented and their conformance with acceptance criteria shall be evaluated by a qualified individual within the responsible organization to ensure that test requirements have been satisfied.

TEST DOCUMENTATION

Test documentation shall include:

- Item or work product tested, date of test, names of tester and data recorders, type of observation and method of testing;
- Identification of test criteria or reference documents used to determine acceptance;
- Results and acceptability of the test;
- Actions taken in connection with any nonconformances or deviations noted;
- Name of the person evaluating the test results; and
- Identification of the measuring and test equipment (M&TE) used during the test.

SECTION 12 CONTROL OF MEASURING AND TEST EQUIPMENT

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 12, Control of Measuring and Test Equipment, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 12 and Supplement 12S-1 of NQA-1-1994 Part I.

This section establishes LES control for tools, gages, instruments and other measuring and test equipment (M&TE) used for activities affecting quality, including design activities where applicable, construction, operation and decommissioning. M&TE is controlled and at specified periods calibrated and adjusted to maintain accuracy within necessary limits. Selection of M&TE shall be controlled to ensure that such items are of proper type, range, accuracy, and tolerance to accomplish the functions of determining conformance to specified requirements.

CALIBRATION

M&TE shall be calibrated, adjusted and maintained at prescribed intervals or, prior to use, against reference calibration standards having traceability to nationally recognized standards. If no nationally recognized standards or physical constants exist, the basis for calibration shall be documented. Calibration standards shall have a greater accuracy than the required accuracy of the M&TE being calibrated. If calibration standards with a greater accuracy than required of the M&TE being calibrated do not exist or are unavailable, calibration standards with accuracy equal to the required calibration accuracy may be used, provided they are shown to be adequate for the requirements. The basis for the calibration acceptance shall be documented and authorized by responsible management as defined in applicable procedures. The level of management authorized to perform this function shall be identified. The method and interval of calibration for each device shall be defined, based on the type of equipment, stability characteristics, required accuracy, intended use and other conditions affecting measurement control. For M&TE used in one- time-only applications, the calibration shall be performed both before and after use. A calibration shall be performed when the accuracy of calibrated M&TE is suspect. Calibrated M&TE shall be labeled, tagged, or otherwise suitably marked or documented to indicate due date or interval of the next calibration and uniquely identified to provide traceability to its calibration data.

DOCUMENTING THE USE OF M&TE

The use of M&TE shall be documented. As appropriate to equipment use and its calibration schedule, the documentation shall identify the processes monitored, data collected or items inspected or tested since the last calibration.

OUT OF CALIBRATION M&TE

M&TE shall be considered to be out-of-calibration and not be used until calibrated if any of the following conditions exist:

- The calibration due date or interval has passed without re-calibration.
- The device produces results known or suspected to be in error.

- Out-of-Calibration M&TE shall be controlled. The controls shall include the following requirements:
    - Out-of-Calibration M&TE shall be tagged, segregated or otherwise controlled to prevent use until they have been recalibrated.

When M&TE is found out-of-calibration, the validity of results obtained using that equipment since its last valid calibration shall be evaluated to verify the acceptability of previously collected data, processes monitored, or items previously inspected or tested. The evaluation shall be documented.

If any M&TE is consistently found out-of-calibration during the re-calibration process, it shall be repaired or replaced.

LOST M&TE

When M&TE is lost, the validity of results obtained using that equipment since its last valid calibration shall be evaluated to determine acceptability of previously collected data, processes monitored or items previously inspected or tested. The evaluation shall be documented.

HANDLING AND STORAGE

M&TE shall be properly handled and stored to maintain accuracy.

COMMERCIAL DEVICES

Calibration and control shall not be required for rulers, tape measures, levels and other normal commercial equipment that provides adequate accuracy.

M&TE DOCUMENTATION

M&TE calibration documentation shall include the following information:

- Identification of the measuring or test equipment calibrated;
- Traceability to the calibration standard used for calibration;
- Calibration data;
- Identification of the individual performing the calibration;
- Identification of the date of calibration and the re-calibration due date or interval, as appropriate;
- Results of the calibration and statement of acceptability;
- Reference to any actions taken in connection with out-of-calibration or nonconforming M&TE including evaluation results, as appropriate; and
- Identification of the implementing document used in performing the calibration.

SECTION 13 HANDLING, STORAGE, AND SHIPPING

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 13, Handling, Storage and Shipping, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 13 and Supplement 13S-1 of NQA-1-1994 Part I.

Handling, storage, cleaning, packaging, shipping and preservation of items are controlled in accordance with requirements of this section to prevent damage or loss and to minimize deterioration.

CONTROLS

Handling, storage, cleaning, packaging, shipping and preservation of items shall be conducted in accordance with established work and inspection implementing procedures, shipping instructions or other specified documents. For critical, sensitive, perishable or high-value articles, specific instructions for handling, storage, cleaning, packaging, shipping and preservation shall be prepared and used.

SPECIAL EQUIPMENT, TOOLS AND ENVIRONMENTS

If required for particular items, special equipment (i.e., containers, shock absorbers and accelerometers) and special protective environments (i.e., inert gas and specific moisture/temperature levels) shall be specified and provided. If special equipment and environments are used, provisions shall be made for their verification. Special handling tools and equipment shall be used and controlled as necessary to ensure safe and adequate handling. Special handling tools and equipment shall be inspected and tested at specified time intervals and in accordance with procedures to verify that the tools and equipment are adequately maintained. Operators of special handling and lifting equipment shall be experienced or trained in the use the equipment.

MARKING AND LABELING

Measures shall be established for marking and labeling for the packaging, shipping, handling and storage of items as necessary to adequately identify, maintain and preserve the item. Markings and labels shall indicate the presence of special environments or the need for special controls if necessary.

SECTION 14 INSPECTION, TEST, AND OPERATING STATUS

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 14, Inspection, Test and Operating Status, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 14 of NQA-1-1994 Part I.

This section establishes requirements for LES to identify the status of inspection and test activities. Status is indicated either on the items or in documents traceable to the items where it is necessary to assure that required inspections and tests are performed and to assure that items which have not passed the required inspections and tests are not inadvertently installed, used or operated. Status is maintained through indicators (i.e., physical location and tags, markings, shop travelers, stamps, inspection records or other suitable means). The authority for application and removal of tags, markings, labels and stamps are specified. Status indicators shall also provide for indicating the operating status of systems and components of the nuclear facility (i.e., tagging valves and switches) to prevent inadvertent operation.

Process control procedures, test and inspection procedures, nonconforming item control procedures, installation records, and checklists are used as applicable to control the installation of structures, system and components. These documents contain hold points, activity checklists, and in many cases, step-by-step signoffs which indicate the status of fabrication, installation, inspections, and test. This system is used to prevent inadvertent use of nonconforming items or bypassing of inspections and tests and prevent inadvertent operation.

During operation, in order to ensure that equipment status is clearly evident, and to prevent inadvertent operation, the LES QA Program requires structures, systems and components that are inoperable to be identified as such. This identification may be by means of tags, labels, stamps or other suitable methods. When tags, labels, or stamps are utilized for the identification of equipment status, the issuance and removal thereof is documented to ensure proper control of such identification measures. Also, procedures require that the operability of an item removed from operation for maintenance or testing be verified prior to returning the item to normal service.

Measures taken by QA personnel, during the performance of required inspection and quality control activities, to identify equipment status are controlled by the QA organization independent of measures taken to identify and control equipment status by LES.

Changing the sequence of inspections, tests, and other activities involving safety requires the same controls as the original review and approval.

SECTION 15 NONCONFORMING ITEMS

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 15, Nonconforming Items, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 15 and Supplement 15S-1 of NQA-1-1994 Part 1.

This section provides the process for controlling items that do not conform to specified requirements. For the purposes of this QAPD, items referenced to in this section means materials, parts, or components. The control of nonconforming activities and services is described in Section 16, Corrective Action. These items are controlled to prevent inadvertent installation or use. The controls provide for identification, documentation, evaluation, segregation when practical, disposition of nonconforming items and for notification to affected organizations.

DOCUMENTING AND EVALUATING NONCONFORMING ITEMS

Nonconformance documentation shall clearly identify and describe the characteristics that do not conform to specified criteria. Nonconformance documentation shall be reviewed by the responsible affected organization and recommended dispositions of nonconforming items shall be proposed in accordance with procedures. The review shall include determining the need for additional corrective actions according to the requirements of Section 16, Corrective Action. In addition, organizations affected by the nonconformance shall be notified. Recommended dispositions shall be evaluated and approved in accordance with procedures. Personnel performing evaluations of recommended dispositions shall have demonstrated competence in the specific area they are evaluating, an adequate understanding of the requirements and access to pertinent background information. The responsibility and authority for reviewing, evaluating, approving the disposition and closing nonconformances shall be specified in procedures. The LES QA Organization is responsible for administering the Nonconformance Process. QA can initiate, recommend, or provide solutions via designated channels. QA will verify the implementation of the corrective actions and QA will assure that procedures are in place to control the installation and use of nonconformances until an acceptable solution has been provided. Further processing, delivery, installation or use of a nonconforming item shall be controlled pending the evaluation and approval of the disposition by authorized personnel.

IDENTIFYING NONCONFORMING ITEMS

Employees of LES and LES contractors have a procedural obligation to identify and document nonconformances. Nonconforming items shall be identified by marking, tagging or other methods that do not adversely affect their end use. The identification shall be legible and easily recognizable. If the identification of a nonconforming item is not practical, the container, package or segregated storage area, as appropriate, shall be identified.

SEGREGATING NONCONFORMING ITEMS

Nonconforming items shall be segregated, when practical, by placing them in a clearly identified and designated hold area until properly dispositioned. If segregation is impractical or impossible due to physical conditions, then other precautions shall be employed to preclude inadvertent use.

DISPOSITION OF NONCONFORMING ITEMS

The disposition, such as “use-as-is,” “reject,” “repair,” or “rework,” of nonconforming items shall be identified and documented. The technical justification for the acceptability of a nonconforming item that has been dispositioned “repair” or “use-as-is” shall be documented.

Items that do not meet original design requirements that are dispositioned “use-as-is” or “repair” shall be subject to design control measures commensurate with those applied to the original design. If changes to the specifying document are required to reflect the as-built condition, the disposition shall require action to change the specifying document to reflect the accepted nonconformance. Any document or record change required by the disposition of the nonconformance shall be identified in the nonconformance documentation; and, when each document or record is changed, the justification for the change shall identify the nonconformance documentation. The disposition of an item to be reworked, or repaired shall contain a requirement to reexamine (inspect, test, or nondestructive examination) the item to verify acceptability. Repaired or reworked items shall be reexamined in accordance with applicable procedures using the original process and acceptance criteria unless the nonconforming item disposition has established alternate acceptance criteria.

TRENDING

Nonconformance documentation shall be periodically analyzed by the LES QA organization to identify adverse quality trends in accordance with Section 16, Corrective Action.

SECTION 16 CORRECTIVE ACTION

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 16, Corrective Action, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 16 of NQA-1-1994 Part 1.

Conditions adverse to quality including activities and services shall be identified promptly and corrected as soon as practical. For significant conditions adverse to quality, the cause of the condition shall be determined and corrective action taken to preclude recurrence. The identification, cause, and corrective action for significant conditions adverse to quality shall be documented and reported to appropriate levels of management. Follow-up action shall be taken to verify implementation of the corrective action. Significant conditions adverse to quality shall be tracked and evaluated so that adverse trends can be identified and appropriate corrective action can be taken.

Procedure(s) shall be issued to establish the CAP which includes the following processes, including closure:

- Prompt identification and correction of conditions adverse to quality;
- Evaluating significant conditions adverse to quality for reportability to the NRC (when required) under 10 CFR 21, Reporting of Defects and Noncompliance, or other applicable reporting requirements and reporting such conditions when warranted;
- Stopping work, if applicable;
- Determining root cause and corrective actions to preclude recurrence for significant conditions adverse to quality; and
- Follow-up actions to verify implementation of corrective actions taken for significant conditions adverse to quality.

IDENTIFYING AND CLASSIFYING CONDITIONS ADVERSE TO QUALITY

Conditions adverse to quality shall be classified in one of two categories in regard to their significance, and corrective actions shall be taken accordingly. The two categories of significance include:

- Conditions adverse to quality
- Significant conditions adverse to quality

Conditions adverse to quality are defined as failures, malfunctions, deficiencies, deviations, defective material and equipment and nonconformances. Conditions adverse to quality shall be documented and reported to the appropriate levels of management.

Responsible management shall investigate and fully identify the condition and document the results. Responsible management shall then utilize investigation results to determine and document corrective action (including remedial action and if appropriate, actions to prevent recurrence). Responsible management shall complete remedial action and document completion of actions in a timely manner.

Significant conditions adverse to quality are defined as:

- A deficiency that would seriously impact an item, activity or service from meeting or

performing its intended function or output of assuring public health and safety;

- A deficiency in design that has been approved for fabrication or construction where the design deviates extensively from design criteria and bases;
- A deficiency in the fabrication or construction of, or significant damage to, structures, systems or components that require extensive evaluation, re-design or repair in order to establish the adequacy of the structure, system or component to perform its intended function of assuring public health and safety;
- A deviation from performance specifications that shall require extensive evaluation, re-design, or repair to establish the adequacy of the structure, system or component to perform its intended function;
- A significant error in a computer program used to support activities affecting quality after it has been released for use;
- A deficiency, repetitive in nature, related to an activity or item subject to the LES QA Program; and
- A condition that, if left uncorrected, has the potential to have a serious negative impact on activities or items subject to the LES QA Program controls.

If a supplier or subtier supplier discovers a defect or noncompliance which the supplier evaluates as a substantial safety hazard, then the supplier shall be required to report the item under 10 CFR 21, Reporting of Defects and Noncompliance, and notify the LES in writing. If the supplier or subtier supplier is unable to determine if the defect/non compliance is a substantial safety hazard then the supplier or subtier supplier is required to report the item to LES for determination of reportability.

Significant conditions adverse to quality shall be evaluated for a stop work condition to determine if stopping work is warranted. If a stop work condition is identified, management shall issue stop work in accordance with the applicable procedure. Upon resolution of the related significant condition adverse to quality, management shall take appropriate action to lift and close (in part or total) the stop work order.

FOLLOW-UP ACTION

The procedure(s) establishing the Corrective Action Program shall include a requirement for management to take follow-up action to verify implementation of corrective action taken to address significant conditions adverse to quality. The QA organization shall be responsible for conducting periodic assessments of these follow-up actions.

TRENDING

The procedure(s) establishing the CAP shall assign organizational responsibility for trending significant conditions adverse to quality and the criteria for determining trends. Reports of significant conditions adverse to quality shall be evaluated to identify adverse quality trends and help identify root causes. Trend evaluation shall be performed in a manner and at a frequency that provides for prompt identification of adverse quality trends. Identified adverse trends shall be handled in accordance with the CAP described here and reported to the appropriate management.

SECTION 17 QUALITY ASSURANCE RECORDS

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 17, Quality Assurance Records, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 17 and Supplement 17S-1 of NQA-1-1994 Part I.

A QA record is any completed record that furnishes documentary evidence of the quality of items and/or activities affecting quality. Records may include specially processed records such as radiographs, photographs, negatives, microforms and magnetic/electronic media. LES completed QA records that furnish documentary evidence of quality shall be specified, prepared and maintained in accordance with applicable regulatory requirements and applicable procedures. QA Records shall be legible, identifiable, retrievable, and shall be protected against damage, deterioration and loss. Requirements and responsibilities for record transmittal, distribution, retention, maintenance and disposition shall be established and documented in procedures. Retention periods for the various types of records generated under the LES QA Program shall be specified as Lifetime or Nonpermanent according to the criteria provided in this Section. The term “records” used throughout this section is to be interpreted as “Quality Assurance Record,” unless otherwise specified.

RECORD MANAGEMENT SYSTEM

LES shall establish a record management system and LES Records Center at the earliest practicable time consistent with the schedule for accomplishing work activities and in compliance with the requirements of this QAPD. The QA records management system shall be defined, implemented and enforced in accordance with written procedures, instructions or other documentation. Records shall be distributed, handled, and controlled in accordance with written procedures.

GENERATION, CLASSIFICATION AND RETENTION OF QA RECORDS

Applicable LES design specifications, procurement documents, test procedures, operational procedures or other documents and procedures shall specify the records to be generated, supplied or maintained. Documents that are designated to become records shall be legible, accurate and completed appropriate to the work accomplished. LES records shall be classified for retention purposes as lifetime records or nonpermanent records in accordance with the criteria provided below.

- Lifetime records are those that meet one or more of the following criteria:
    - Those which would be of significant value in demonstrating capability for safe operation;
    - Those which would be of significant value in maintaining, reworking, repairing, replacing or modifying an item;
    - Those which would be of significant value in determining the cause of an accident or malfunction of an item; and/or
    - Those which provide required baseline data for in-service inspections.

Lifetime records are required to be maintained for the life of the particular item while it is installed in the facility or stored for future use.

Nonpermanent records are those required to show evidence that an activity was performed in accordance with the applicable requirements of the LES QA Program but need not be retained for the life of the item because they do not meet the criteria for lifetime records. The retention period for nonpermanent records shall be documented in the applicable procedure.

Procedures shall identify those documents that will become QA records. The individual using the procedure is responsible for ensuring the QA records required by the procedure are submitted to the LES Records Center. Documents that may become records shall be maintained and processed in a prudent manner to avoid unnecessary delay and/or expense in retrieving the record when the record is needed to support other work.

Individuals creating records shall ensure the records are legible, accurate and complete, and shall protect them from damage, deterioration or loss during the time the records are in their possession.

Documents shall be considered valid records only if authenticated (i.e., stamped, initialed or signed and dated complete by authorized personnel). If the nature of the record precludes stamping or signing, then other means of authentication by authorized personnel is permitted. This may take the form of a statement by the responsible individual or organization. Handwritten signatures are not required if the document is clearly identified as a statement by the reporting individual or organization. QA records may be originals or copies. LES contractors shall submit to the LES Records Center those records being temporarily stored by them in accordance with contractual requirements. The timing of the submittal shall be as records become completed, or as items are released for shipment, or as prescribed by QA procedures and procurement documents. Records shall be controlled and submitted to the records management system in accordance with implementing procedures.

RECEIVING QA RECORDS

Each organization responsible for receiving records shall provide protection from damage or loss during the time that the records are in their possession. A receipt control system shall be established by the organization to include the following:

- A method for designating the required records;
- A method for identifying records received;
- Procedures for receipt and inspection of incoming records; and
- A method for submittal of completed records to the storage facility without unnecessary delay; and
- Capability to provide current and accurate status of records during the receipt process.

Records shall be indexed to ensure retrievability. Records and/or indexing systems shall provide sufficient information to permit identification between the record and the item or activity to which it applies. The indexing system shall include:

- The location of the records within the records management system;
- Identification of the item or related activity to which the records pertain; and
- The retention classification of the record.

STORING, SAFEKEEPING, AND PRESERVING QA RECORDS

Records shall be stored and preserved in the LES Records Center in accordance with a procedure that includes the following:

- Assignment of responsibility for enforcing the requirements of the procedure;
- A description of the storage facility;
- A description of the filing system to be used;
- A method for verifying that the records received are in agreement with the transmittal document;
- A method for verifying that the records are those designated and the records are legible and complete;
- A description of rules governing control of the records, including access, retrieval and removal;
- A method for maintaining control of and accountability for records removed from the storage facility;
- A method for filing supplemental information and disposition of superseded records;
- A method for precluding entry of unauthorized personnel into the storage area to guard against larceny and vandalism; and
- A method for providing for replacement, restoration or substitution of lost or damaged records.

Storage methods shall be approved by the organization responsible for storage to preclude deterioration of records in accordance with the following:

- Provisions shall be made in the storage arrangement to prevent damage from moisture, temperature and pressure.
- Approved filing methods shall require records to be firmly attached in binders, or placed in folders or envelopes, for storage in steel file cabinets or on shelving in containers appropriate for the record medium being stored.
- The storage arrangement shall provide adequate protection of special processed records (e.g., radiographs, photographs, negatives, microform and magnetic media) to prevent damage from humidity, temperature, excessive light, electromagnetic fields or stacking, consistent with the type of record being stored.

LES RECORDS CENTERS

Originating organizations shall store records in temporary storage while active and required for use; subsequently the records shall be transmitted for permanent storage in accordance with the requirements of this Section and associated procedures.

LES organizations shall provide for temporary storage of records during processing, review or use, until turnover to the LES Records Center for disposition, according to implementing procedures and the following requirements:

- Records shall be temporarily stored in a container or facility with a fire rating of one (1) hour. The temporary storage container or facility shall bear an Underwriters' Laboratories label (UL) (or equivalent) certifying one (1) hour fire protection, or be certified by a person

competent in the technical field of fire protection.

- The maximum time limit for keeping records in temporary storage shall be specified by implementing procedures consistent with the nature or scope of work.

LES QA records permanent storage shall either invoke the alternate single storage facility provision of Section 4.4.2 and/or the dual facilities provision of Section 4.4.4 of Supplement 17S-1 of NQA-1-1994. With either provision used, the LES Records Center shall be constructed and maintained in a manner that minimizes the risk of damage or destruction from the following:

- Natural disasters (i.e., winds, floods or fires);
- Environmental conditions (i.e., high and low temperatures and humidity); and
- Infestation of insects, mold or rodents.

If the alternate single storage facility provision is used, then LES records shall be stored in the LES Records Center in two (2) hour fire rated Class B file containers meeting the requirements of National Fire Protection Association (NFPA) 232-1986 or NFPA 232AM-1986 or both.

If the dual storage facility provision is used for hard copies, then LES records shall be stored with one copy in the LES Records Center and the second copy stored in facility that is sufficiently remote from the Records Center to eliminate the chance of exposure to a simultaneous hazard. If the dual storage facilities provision is used via scanned documents into an electronic records management system, then a back-up tape shall be periodically made of the electronic records management system and its contents and the tape shall be stored in atemporary storage devise in a fire-proof safe. This process invokes the dual storage provision as one copy resides on the records management system computer and a second copy of the total records system resides in a remote location with temporary storage being used for records entered in the interim.

RETRIEVING AND DISPOSITIONING QA RECORDS

The records management system shall provide for retrieval of records in accordance with planned retrieval times based upon the designated record type. Access to records storage facilities shall be controlled. A list shall be maintained designating personnel who are permitted access to the records at the LES Records Center.

Records maintained by a supplier at its facility or other location shall be accessible to the purchaser or designated alternate. The supplier's records shall not be disposed of until contractual requirements are satisfied.

Records accumulated at various locations prior to transfer shall be made accessible to LES directly or through the procuring organization. The record-keeper shall inventory the submittals, acknowledge receipt and process these records in accordance with this QAPD. Various regulatory agencies have requirements concerning records that are within the scope of this Section. The most stringent requirements should be used in determining the final disposition. The supplier's nonpermanent records shall not be disposed of until the applicable conditions listed below are satisfied.

- Items are released for shipment, a Code Data Report is signed, or a Code Symbol stamp is affixed.
- Regulatory requirements are satisfied.

- Operational status permits.
- Warranty consideration is satisfied.
- Purchaser's requirements are satisfied.

RETENTION OF QA RECORDS

Lifetime records shall be retained and preserved for the operating life of the particular item while it is installed in the plant or stored for future use. Nonpermanent records shall not be disposed of until the following conditions are met:

- Regulatory requirements are satisfied;
- Facility status allows document disposal; and
- LES QAPD requirements are satisfied

CORRECTING INFORMATION IN QA RECORDS

Corrections shall include the identification of the person authorized to make the correction and the date the correction was made. Corrections to records shall be performed in accordance with implementing procedures, which provide for appropriate review or approval of the corrections, by the originating organization.

REPLACING LOST OR DAMAGED QA RECORDS

Replacement, restoration or substitution of lost or damaged records shall be performed in accordance with implementing procedures, which provide for appropriate review or approval by the originating organization and any additional information associated with the replacement.

SECTION 18 AUDITS

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 18, Audits, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 18 and Supplement 18S-1 of NQA-1-1994 Part 1.

In accordance with the description of the QA organization during the various phases of design, construction, and operation provided in Section 1, Organization, the LES QA Director or QA Manager shall verify LES compliance with all aspects of the LES QA Program and determine QA Program effectiveness by ensuring that planned and scheduled audits are conducted. Elements that have been selected for audit shall be evaluated against specified requirements. An auditing function reports to the LES QA Director/QA Manager and has the organizational independence and authority to execute an effective audit process to meet all requirements of the QA Program. Objective evidence shall be examined to the depth necessary to determine if these elements are being implemented effectively. LES audits are performed in accordance with written procedures or checklists by appropriately trained and qualified personnel who do not have direct responsibility for performing the activities being audited. Audit results are documented and provided to the appropriate management for review and corrective action as applicable. Follow-up actions are taken where indicated.

AUDIT SCHEDULES

Internal or external audits shall be scheduled in a manner to provide coverage, consistency and coordination with ongoing work, and at a frequency commensurate with the status and importance of the work. Internal or external audits shall be scheduled to begin as early in the life of the work as practical and shall be scheduled to continue at intervals consistent with the schedule for accomplishing the work. As a minimum, internal audits of LES QA Level 1 activities shall be at least once per year or at least once during the life of the activity, whichever is shorter. Regularly scheduled internal audits shall be supplemented by additional audits of specific subjects when necessary to provide an adequate assessment of compliance or effectiveness. Internal audits to determine quality assurance program effectiveness shall be performed on selected work products. The audit schedule shall be developed annually and revised as necessary to ensure that coverage is maintained current. Frequency of audits should be based upon evaluation of all applicable and active elements of the LES QAPD applicable to LES workscope. These evaluations should include an assessment of the effectiveness of the applicable and active elements of the LES QAPD based upon previous audit results and corrective actions, nonconformance reports, identified trends, and significant organizational changes.

AUDIT PLANS

A documented audit plan shall be developed for each audit. This plan shall identify the audit scope, requirements for performing the audit, type of audit personnel needed, work to be audited, organizations to be notified, applicable documents, audit schedule, and implementing documents or checklists to be used.

AUDIT TEAMS

The LES QA Director or QA Manager shall select and assign auditors who are independent of any direct responsibility for performing the work being audited. Audit personnel shall have sufficient authority and organizational freedom to make the audit process meaningful and effective. The audit team shall include one or more auditors comprised of representatives from the LES QA organization and any applicable technical organizations. A lead auditor shall be appointed to supervise the team, organize and direct the audit, prepare and coordinate issuance of the audit report and evaluate responses. Technical specialists may be used to assist in assessing the adequacy of technical processes. Before commencing the audit, the lead auditor shall ensure the personnel assigned to the audit team are prepared and collectively have experience and/or training commensurate with the scope, complexity or special nature of the work to be audited. Lead auditors, auditors and technical specialists shall be trained and qualified according to the requirements of Section 2, Quality Assurance Program.

PERFORMING AUDITS

The LES QA Director or QA Manager shall provide written notification of a planned audit to the affected organizations at a reasonable time before the audit is to be performed. The notification should include all relevant information pertaining to the audit, such as schedule, scope and names of audit lead and team members, if known. In addition, the audit team leader shall ensure the following is performed.

- The audit team shall be adequately prepared before starting the audit.
- Audits shall be performed in accordance with written procedures or checklists.
- Elements that have been selected for the audit shall be evaluated against specified requirements.
- Objective evidence shall be examined to the depth necessary to determine if the selected elements are being implemented effectively.
- Audit results shall be documented by auditing personnel, and reported to/reviewed by management having responsibility for the area audited. Conditions requiring prompt corrective action shall be reported immediately to management of the audited organization.
- Identified audit findings shall be documented and the audited organization shall correct the findings according to the requirements of Section 16, Corrective Action. Minor audit findings can be corrected during the conduct of the audit.

REPORTING AUDIT RESULTS

The audit report shall be prepared and signed by the audit team leader and issued to the management of the audited organization in a timely manner after completion of the audit.

The audit report shall include the following information:

- A description of the audit scope.
- Identification of the auditors.
- Identification of persons contacted during the audit.
- A summary of audit results and the documents reviewed, persons interviewed and the specific results of the reviews and interviews (i.e., a summary of the checklist contents).

- Statement as to the effectiveness of the implementation of the QA Program elements audited.
- A description of each reported adverse audit finding in sufficient detail to enable corrective action to be taken by the audited organization.
- A requested date for response by the audited organization.

RESPONDING TO AUDITS

Management of the audited organization or activity shall:

- Investigate adverse audit findings in a timely manner;
- Determine and schedule corrective action, including measures to prevent recurrence;
- Prior to or by the requested response date, notify the LES QA Director in writing of the actions taken or scheduled, according to the requirements of Section 16 Corrective Action.

EVALUATING AUDIT RESPONSES

The LES QA Director or QA Manager is responsible for evaluating audit responses.

FOLLOW-UP ACTION

Follow-up action shall be taken by the LES QA Director to verify that:

- Corrective actions are completed as scheduled according to the requirements of Section 16 Corrective Action.

RECORDS

- Audit records include audit plans and audit reports.
- Written replies and the record of completion of any required corrective actions.

These documents are QA records and shall be submitted to the LES Records Center for retention according to the requirements of Section 17, Quality Assurance Records.

NON-LES AUDITOR QUALIFICATIONS

Non-LES certified auditors may be used to perform audits and surveillances provided the LES QA Director or QA Manager confirms and documents applicable QAPD requirements have been met and the individual has been certified in accordance with the QA procedure on auditor qualification and certification.

SECTION 19 PROVISIONS FOR CHANGE

This QAPD is reviewed and revised as necessary to reflect any changes that occur during the design, construction, operation, including maintenance and modifications, and decommissioning phases. In addition, this QAPD is revised when corrective actions, regulatory, organizational, or work scope changes warrant changes to the LES QA Program. The LES QAPD is maintained current through design, construction, operation and decommissioning of the facility. The LES QAPD is kept current as the design, construction, operation, and decommissioning activities progress, and appropriate changes are made based on any of the following:

- Lessons learned from audit and assessment findings,
- Program improvements identified from analysis of trends, and
- Changes due to regulations, commitments, reorganizations, revised project schedule, or program improvements from continuous review of assessment results and process improvement initiatives.

Changes to the LES QA Program shall be incorporated in this QAPD and submitted to the NRC within 30 days of implementation prior to and after NRC issuance of the License. Any changes that reduce commitments in the approved QAPD, including those commitments that address the safety program and integrated safety analysis regulatory requirements, as well as the QA Level requirements in this QAPD, will be submitted to the NRC for review and approval prior to implementation.

SECTION 20 QUALITY ASSURANCE PROGRAM FOR QA LEVEL 2 ACTIVITIES

This section outlines the owner defined Quality Assurance Program for QA Level 2 activities. For contractors, the QA Level 2 program shall be described in documents that must be approved by LES. The QA Level 2 program shall be applied to owner designated structures, systems, components, and activities. An International Organization for Standardization (ISO) 9000 series QA program is acceptable for QA Level 2 applications provided it complies with LES QAPD requirements and the ISO program is reviewed and approved by the LES QA Director.

Requirements for QA Level 2 are defined below. QA Level 2 requirements shall not be applied to IROFS or items that may affect the functions of the IROFS.

ORGANIZATION

The organization, lines of responsibility and authority are clearly established and documented.

PERSONNEL QUALIFICATIONS

Measures are established to provide for indoctrination and training of personnel to ensure suitable proficiency is achieved and maintained. Where specific qualifications are required by codes and standards, measures shall be taken to document the qualifications.

PROCEDURES

Work activities are performed in accordance with written procedures. Procedures shall contain the appropriate criteria for determining that prescribed activities have been satisfactorily accomplished.

DOCUMENT CONTROL

Procedures are established to ensure that appropriate documents are properly initiated, changed, and controlled to prevent use of incorrect or superseded documents.

DESIGN CONTROL

The design shall be defined, controlled, and verified. Applicable design inputs shall be appropriately specified on a timely basis and correctly translated into design documents. Design interfaces are identified and controlled. Design adequacy is verified by persons independent of those who performed the design. Design changes are governed by control measures commensurate with those applied to the original design. Design of systems, structures or components may be verified by the development and service testing of hardware similar to the equipment to be used in the facility. Installation and use of this type of equipment requires approval of LES management.

CONTROL OF PURCHASED ITEMS AND SERVICES

Measures are established to ensure conformance with the specified requirements. Measures are established to ensure suppliers of materials, equipment, or services are capable of supplying these items to the quality specified in the procurement documents. This may be done

by evaluation and approval of the supplier's products and facilities or audits of the supplier's quality program.

CONTROL OF PROCESSES, MEASURING AND TEST EQUIPMENT

Processes affecting quality of items or services are controlled. Special processes such as welding, heat treating, and nondestructive examination shall be performed by certified personnel using certified procedures in accordance with specified requirements. To maintain accuracy within specified limits, the LES QA Program requires that devices (e.g., tools, gauges, instruments), and measuring and test equipment including process-related instrumentation and controls that are used in activities affecting the quality of items, are properly controlled, calibrated, and adjusted at specified periods in accordance with written procedures.

INSPECTIONS

Inspections required to verify conformance of an item or activity to specified requirements are planned and executed. Characteristics to be inspected and inspection methods to be employed are specified. Inspection results are documented. Inspections for acceptance are performed by persons other than those who performed the work being inspected.

NONCONFORMANCES AND CORRECTIVE ACTION

Measures are established so conditions adverse to required quality are promptly identified and corrected. Controls are established to prevent inadvertent installation or use of items that do not conform to specified requirements.

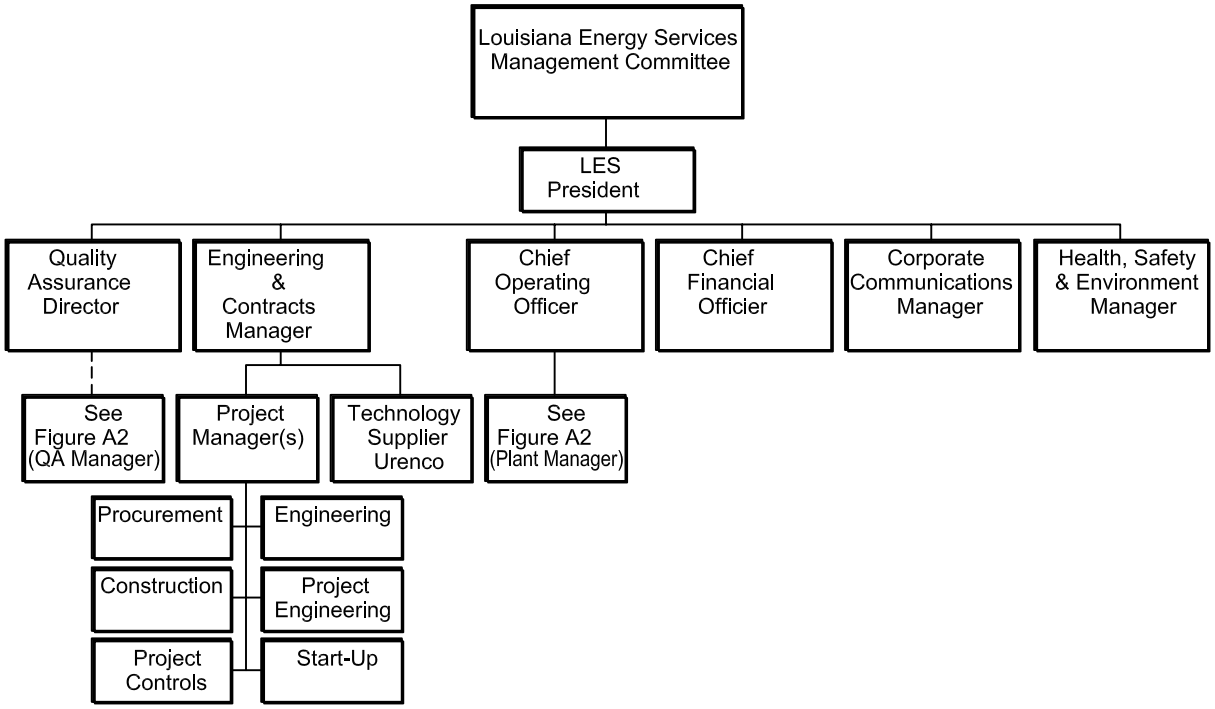
RECORDS

Records that furnish documentary evidence of quality are specified, prepared, and maintained. Records shall be legible, identifiable, and retrievable. Records are protected against damage, deterioration, and loss. Requirements and responsibilities for record transmittal, distribution, retention, maintenance, and disposition are established and documented.

AUDITS AND ASSESSMENTS

Measures are established to verify compliance with the LES QA Program and to determine its effectiveness. The results are documented and reported to and reviewed by responsible management. Follow-up action shall be taken where indicated.

FIGURES



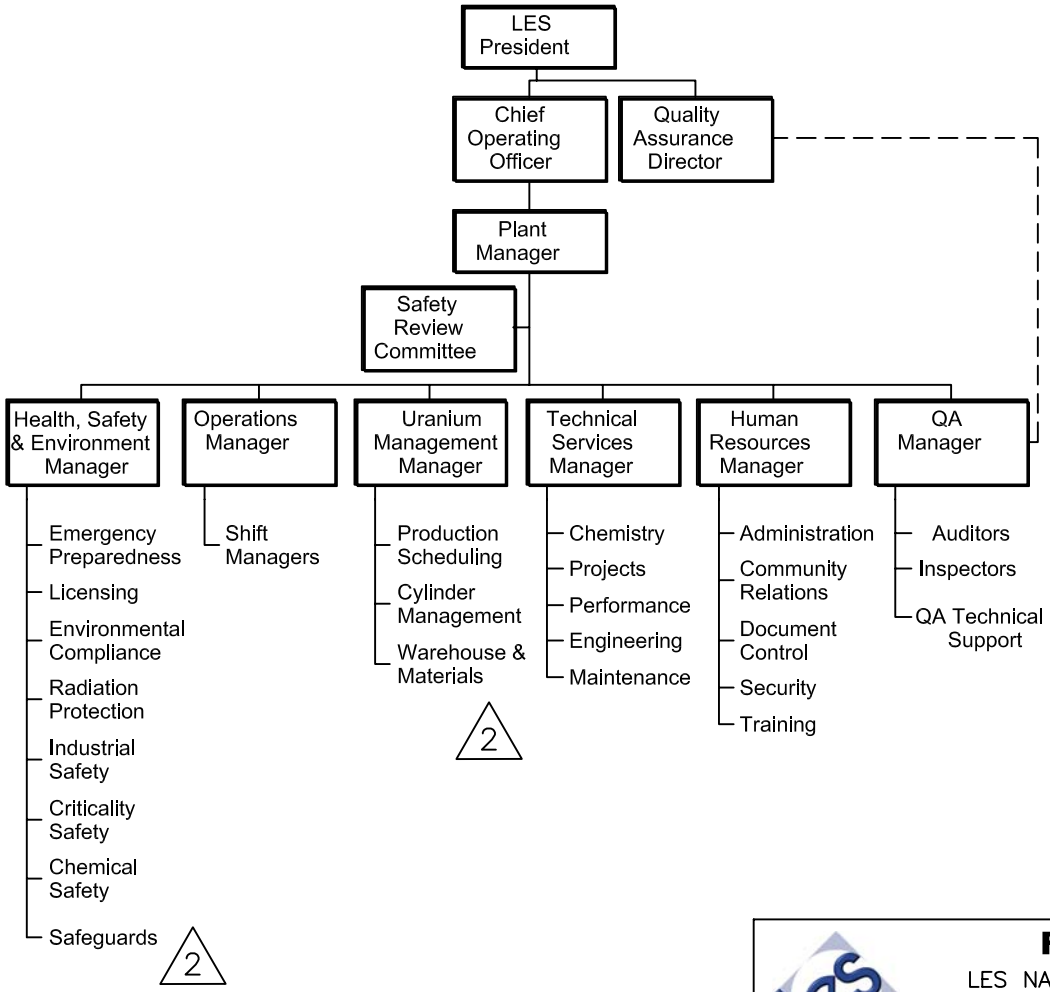
2

REFERENCE NUMBER
Figure A1.dwg

FIGURE A1

LES CORPORATE, DESIGN AND CONSTRUCTION ORGANIZATION

REVISION 2 DATE: JULY 2004



REFERENCE NUMBER
Figure A2.dwg

FIGURE A2

LES NATIONAL ENRICHMENT PLAN
OPERATING ORGANIZATION

REVISION 2 DATE: JULY 2004