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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 20.110, 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 will use, 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).

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), 2.7mSv (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.

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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. 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) and 8.29 (NRC, 1996). 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, (4) and 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.

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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). A 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. At the minimum, the RWP requires approval by a staff member who is a radiation specialist. Radiation Work Permits (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.

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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.

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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.

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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.

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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.

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

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TABLES

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Table 4.1-1 Administrative Radiation Exposure Limits

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	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

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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

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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

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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%

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FIGURES

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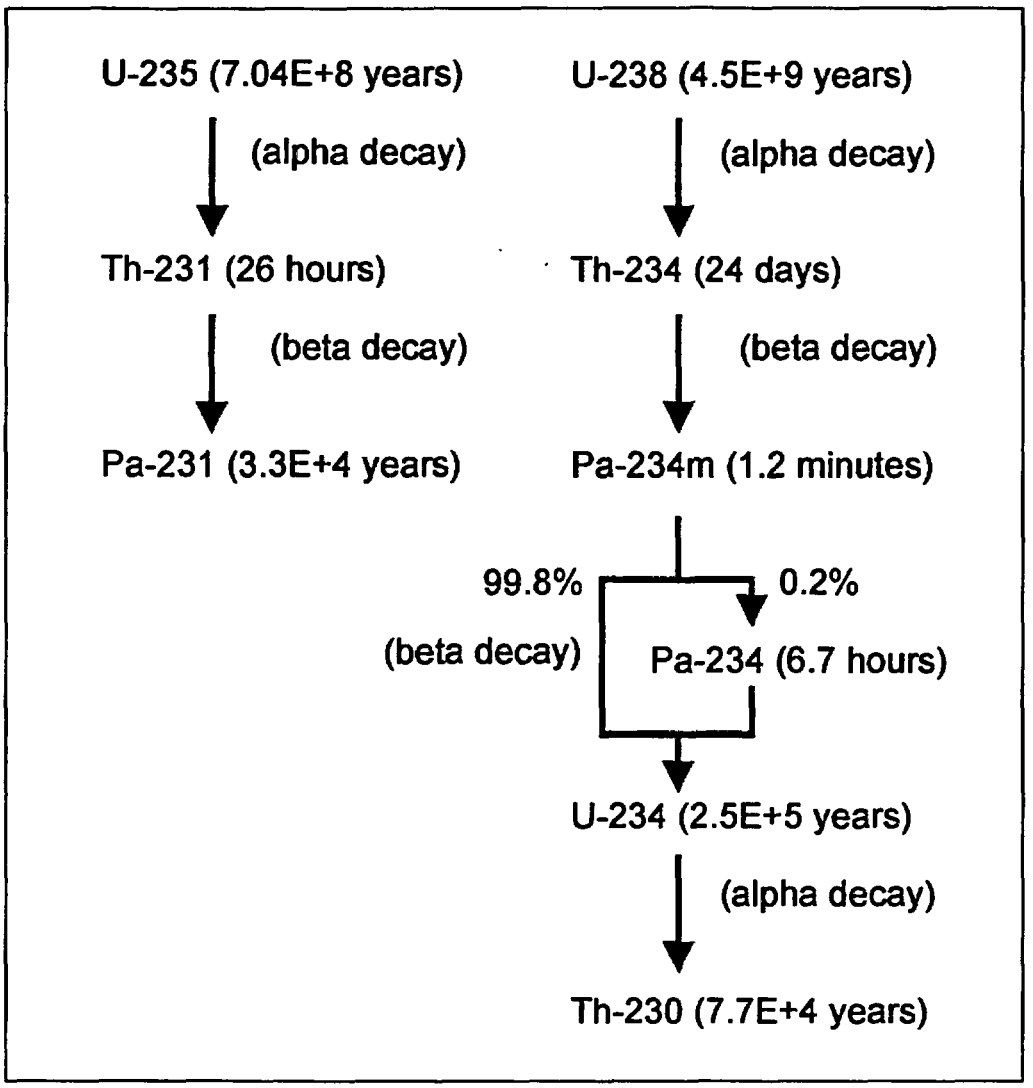

FIGURE 4.7-1
URANIUM AND DECAY PRODUCTS OF INTEREST

REFERENCE NUMBER
Figure 4.7-1.doc

REVISION DATE: DECEMBER 2003

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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 follows the guidelines in this regulatory guide for specific ANSI/ANS criticality safety standards.

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
Management Measures	70.62	5.4.3.3
Section 5.2 Methodologies and Technical Practices		
Methodology	70.61	5.4.3.4.1
Section 5.3 Nuclear Criticality Safety Determinations		
Determining the criticality of various systems and configurations	70.61(d)	5.4.3.4.4
Section 5.4 Technical Practices		
Criticality Prevention by Engineered Controls	70.61(d)	5.4.3.4.2
Criticality Prevention by Administrative Controls	70.61	5.4.3.4.2
5.5 Criticality Accident Alarm System (CAAS)		
Criticality Accident Alarm System	70.24	5.4.3.4.3
5.6 Criticality IROFS		
Criticality IROFS	70.64(a)	5.4.3.4.5
5.7 Additional Nuclear Criticality Safety Program Commitments		
Additional NCS Program Commitments	70.65(b)	5.4.3.4.6 5.4.3.4.7

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 5.6, Criticality IROFS.

5.1.1 Management of the Nuclear Criticality Safety (NCS) Program

The NCS criteria in Sections 5.3.16, Additional NCS Determinations, and 5.7, Additional NCS Program Commitments, are used for managing criticality safety include adopting the double contingency principle as stated in the ANSI/ANS-8.1-1983, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors (ANSI, 1983a). The adopted double contingency principle states "process design shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." In the current design each process that has accident sequences that could result in an inadvertent nuclear criticality at the NEF will have double contingency protection. In most cases double contingency protection will be 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 $^{w}/_{o}$ 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 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 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.

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.

5.1.2 Control Methods for Prevention of Criticality

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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. The criticality safety staff 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)
- Perform criticality safety 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. A criticality safety 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)-8.1-1983, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI, 1983a). 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.

5.1.6 Management Measures

Chapter 11, Management Measures, describes the management measures applied to IROFS to ensure that the IROFS are available and able to perform their functions when needed. Management measures include training and qualifications, procedures, configuration management, records management and audits and assessments. Specific criticality-related management measures are discussed in the following sections.

Additionally, a formal configuration management program is implemented (see Section 11.1, Configuration Management (CM)). This program ensures that the facility design remains consistent with the design analyzed by the NCSEs. The program also ensures that changes to the facility design have the appropriate review and controls in place. The implementation of this formal configuration management program ensures that: (1) facility changes are managed to maintain the integrity of the safety basis and to ensure the changes receive the appropriate level of criticality safety review, and (2) changes requiring NRC approval are appropriately identified and treated. Louisiana Energy Services (LES) will implement measures to meet the requirements of 10 CFR 70.64 (CFR, 2003c) to ensure that the facility design meets the baseline design criteria for criticality safety as described in Table 5.1-2, Safety Criteria for Buildings/Systems/Components.

5.1.6.1 Nuclear Safety Training

Employees must complete formal nuclear safety training prior to being granted unescorted access in the Controlled Access Area. Methods for evaluating training effectiveness include an initial examination covering the formal training content and observations of operational activities as appropriate during scheduled audits and inspections.

Trained instructors are approved by the criticality safety organization. The instructors ensure that the content of the training program is current and adequate by reviewing the training program content on a regularly scheduled basis.

Records of previously trained employees who are allowed unescorted access to the NEF are retained in accordance with the records management system. Visitors are trained commensurate with the scope of their visit or are escorted by trained employees. Nuclear

criticality safety training includes training on the following subjects, as applicable to the functions performed:

- Use of process parameters credited for nuclear criticality safety control
- Nuclear criticality safety postings that identify administrative controls for operators
- Fission chain reactions and accident consequences
- Neutron behavior in a fissioning system
- IROFS for criticality safety
- Selected criticality accident histories
- Response to criticality accident alarm system (CAAS) signals
- Policies and procedures
- Instructions on reporting defective NCS conditions.

The above training will be in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996) and ANSI/ANS-8.20-1991 (ANSI, 1991) as they relate to training.

5.1.6.2 Criticality and Radiation Assessments

Representatives of the criticality safety and radiation protection organizations will conduct formal, scheduled safety assessments in accordance with documented, approved procedures. These assessments will ensure that operations conform to criticality and radiation requirements in accordance with ANSI/ANS-8.19-1996, Administrative Practices for Nuclear Criticality Safety (ANSI, 1996).

Criticality and radiological assessments are performed under the direction of the criticality safety staff and the Radiation Protection Manager. Personnel performing these assessments do not report to the production organization and have no direct responsibility for the function and area being assessed.

Assessment results are communicated in writing to the Operations Manager and to the HS&E Manager. Required corrective actions are documented and approved by the HS&E Manager and are reported to the Plant Manager.

The Operations Group is assessed periodically to ensure that procedures are being followed and that 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. These assessments are conducted, in consultation with operating personnel, by NEF staff who are knowledgeable in nuclear criticality safety and who are not immediately responsible for operations.

Weekly nuclear criticality safety walkthroughs of UF_6 process areas are conducted and documented. Identified weaknesses are entered into the facility corrective action program and are promptly resolved.

5.1.6.3 Independent Audits

Appropriately trained and experienced individuals who have independence from the organization and who are not involved in the routine performance of the work or program being audited audit the Radiation Protection and Nuclear Criticality Safety Programs on a planned, scheduled basis. The scope of independent audits covers the adequacy of the safety program, as well as compliance with requirements. NCS audits are conducted and documented quarterly such that all NCS aspects of the management program will be audited at least every two years. The Quality Assurance (QA) Department provides the lead for managing the audits utilizing the technical expertise from the line organization.

Audit results are reported in writing to the Plant Manager, the Operations Manager, and the HS&E Manager.

5.1.6.4 Nuclear Criticality Safety Procedures

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 management procedure to ensure that they are maintained current.

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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 (AEA, 1998) 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 (AEA, 1998) 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 (AEA, 1998) 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 (AEA, 1998) 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). 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 1.0016 ± 0.0005 and the results are shown in Figure 5.2-1,Validation Results for Uranium Solutions, plotted against H/U-fissile ratio. If only the 52 low-enriched solutions are considered, the mean calculated value is 1.0008 ± 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 will be performed. LES will complete the validation of the MONK8A code for the specific criticality analyses described in this chapter. Specifically, the experiments provided in Table 5.2-1, Uranium Solution Experiments Used for Validation, will be re-run and documented in the integrated safety analysis for the National Enrichment Facility. Since MONK8A has been extensively validated and verified in past experiments, no appreciable change in the results is anticipated.

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, while the facility is limited to an enrichment of 5.0 $^{w}/_{o}$ ^{235}U. Details of the criticality safety analyses are provided in Section 5.3, NCS Determinations.

5.2.1.3 General Nuclear Criticality Safety Methodology

The nuclear criticality safety determinations presented in Section 5.3, NCS Determinations, 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, NCS Determinations). 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 (AEA, 1998) 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, NCS Determinations.

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 (AEA, 1998). This adequately covers all models likely to be purchased.
- The UF_6 pumping units are a combination unit of two pumps, one 500 m^3/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 m^3/hr (70,626 ft^3/hr) pump which is modeled explicitly according to manufacturer's drawings.

5.2.1.4 Nuclear Criticality Safety Analysis

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

- NCS determinations will be performed using acceptable methodologies.
- Methods will be validated and used only within demonstrated acceptable ranges.
- The analyses will adhere to ANSI/ANS-8.1-1983 (ANSI, 1983a) 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 (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 determinations will be performed in accordance with the methods specified and incorporated in the management program.

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5.3 NUCLEAR CRITICALITY SAFETY DETERMINATIONS

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5.4 TECHNICAL PRACTICES

The NEF will be designed and operated based on the following NCS criteria, listed in the order of priority:

- Passive control (e.g., use of safe geometry)
- Active control (e.g., use of engineered control)
- Administrative control (e.g., mass control; moderator control).

5.4.1 Criticality Prevention by Passive Control

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5.4.3 Criticality Prevention by Administrative Controls

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5.4.4 Safety Review Committee

The NEF maintains a Safety Review Committee to assist with the safe operation of the facility. The Safety Review Committee (SRC) reports to the Plant Manager, and provides technical and administrative review and audit of facility operations, which could impact plant worker and public safety. The scope of activities reviewed and audited by the SRC includes the following:

- Radiation protection
- Nuclear criticality safety control
- Hazardous chemical safety
- Industrial safety including fire protection
- Environmental protection
- As low as reasonably achievable (ALARA) policy implementation
- Changes in facility design or operations.

See Chapter 2, Organization and Administration, for additional information on the Safety Review Committee.

5.4.5 Audits and Assessments

Audits and assessments are conducted to determine that plant operations are performed in compliance with regulatory requirements, license conditions, and written procedures for activities related to criticality safety control.

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.

Qualified staff personnel that are not directly responsible for production activities being inspected perform assessments routinely. Assessments are conducted at least semi-annually. Deficiencies noted during the assessment requiring corrective action are entered into the corrective action program and forwarded to the manager of the applicable area or function for action. Future assessments include a review to evaluate if corrective actions have been effective.

The SRC and the Quality Assurance Department are responsible for audits. SRC audits apply to the areas described in Section 5.3, NCS Determinations. Quality Assurance audits apply to activities subject to the LES QA Program.

The results of the audit are provided in a written report within 30 days of the audit to the Plant Manager, the SRC, and the manager responsible for the activities audited. The manager or designee responds to any deficiencies identified in the audits within 30 days. Open deficiencies are tracked to completion by a designated member of the audit organization (SRC or QA), and re-examined during future audits or assessments to ensure corrective action has been completed.

See Section 11.5, Audits and Assessments, for additional information.

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5.5 CRITICALITY ACCIDENT ALARM SYSTEM (CAAS)

The facility will be 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 will be provided with CAAS coverage.

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5.6 CRITICALITY ITEMS RELIED ON FOR SAFETY

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5.7 ADDITIONAL NUCLEAR CRITICALITY SAFETY PROGRAM COMMITMENTS

The following are additional commitments of the NCS Program:

- The double contingency principle will be used in determining NCS controls and IROFS in the design of new facilities or new processes at the NEF that require a license amendment under 10 CFR 70.72 (CFR, 2003e). The double contingency protection as discussed in NUREG-1520 (NRC, 2002) Section 5.4.3.4.4(9) will be used.

- The acceptance criteria in NUREG-1520 (NRC, 2002) Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences will be met.

- Appendix A of ANSI/ANS-8.1-1983 (ANSI, 1983a) will be used in determining NCS accident sequences for the NEF.

- ANSI/ANS-8.10-1983 (ANSI, 1983b), as modified by Regulatory Guide 3.71 (NRC, 1998), will be used in determining the consequences of NCS accident sequences.

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

- The facility change mechanism process will be supported by performing NCS determinations to evaluate changes to processes, operating procedures, IROFS, and management measures.

- The NCS program will be upgraded to reflect changes in the ISA or new NCS methodologies and to modify operating and maintenance procedures in ways that could reduce the likelihood of occurrence of an inadvertent nuclear criticality.

- Records of NCS programs will be retained and any corrective actions taken will be documented.

- 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.

- A change control process will be used that is sufficient to ensure that the safety basis of the facility will be maintained during the lifetime of the facility. The change process will be documented in written procedures and will ensure that all potentially affected SNM 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 change control

process will have procedures for the review and approval of facility changes by the NCS criticality engineer to determine the potential effects on NCS.

- The change control process will be connected to the facility's configuration management system to ensure that changes to the NCS basis are incorporated into procedures, evaluations, postings, drawings, other safety basis documentation, and the ISA Summary.

- A program to determine whether facility changes require NRC approval in accordance with the 10 CFR 70.72(c) (CFR, 2003e) will be provided. This program will be documented in written procedures and will involve individuals qualified to determine the incremental effect of changes to the safety basis as documented in the ISA Summary. All proposed changes will be compared to the approved ISA Summary.

- 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 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.

5.8 REFERENCES

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TABLES

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

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

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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)
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
23	Uranyl nitrate solution (~ 95 $^{w}/_{o}$ enriched)	5	HEU-SOL-THERM-013
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

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FIGURES

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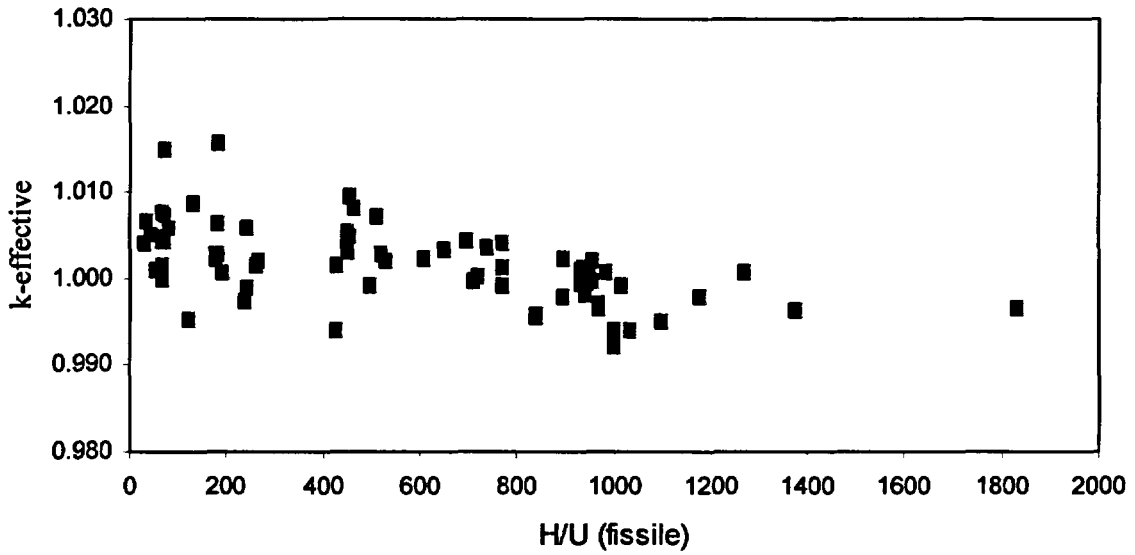

FIGURE 5.2-1
VALIDATION RESULTS FOR URANIUM SOLUTIONS

REFERENCE NUMBER
Figure 5.2-1.doc

REVISION DATE: DECEMBER 2003

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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

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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.

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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.

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

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.

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

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 (FR, 2002) for normalizing chemical and radiological exposure to values appropriate for the time intervals under consideration. The rationale associated with exposure times are further defined below.

6.3.2.1.1 Worker Exposure Assumptions

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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).

6.4.2.1.1 Physical Barriers

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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

The NEF has a facility emergency plan and program which includes response to mitigate the potential impact of any process chemical release including requirements for notification and reporting of accidental chemical releases.

A more detailed description of the emergency response program can be found in the NEF, Emergency Plan.

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.

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TABLES

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Table 6.1-6 Physical Properties of UF_6

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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

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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

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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

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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
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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	1.6	1.6	1.6	0.8	0.8
ERPG-2	16.4	AEGL-2	78	28	20	9.8	7.0
ERPG-3	41	AEGL-3	139	51	36	18	12

Table 6.3-4 ERPG and AEGL values for Uranium Hexafluoride (as soluble U)
Page 1 of 1

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	374	68	24	3.0	1.1

Table 6.3-5 Enhanced Definition of Consequence Severity Categories

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		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) (5-min exposure)	>1,075 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; >1.6 mg HF/m^3

FIGURES

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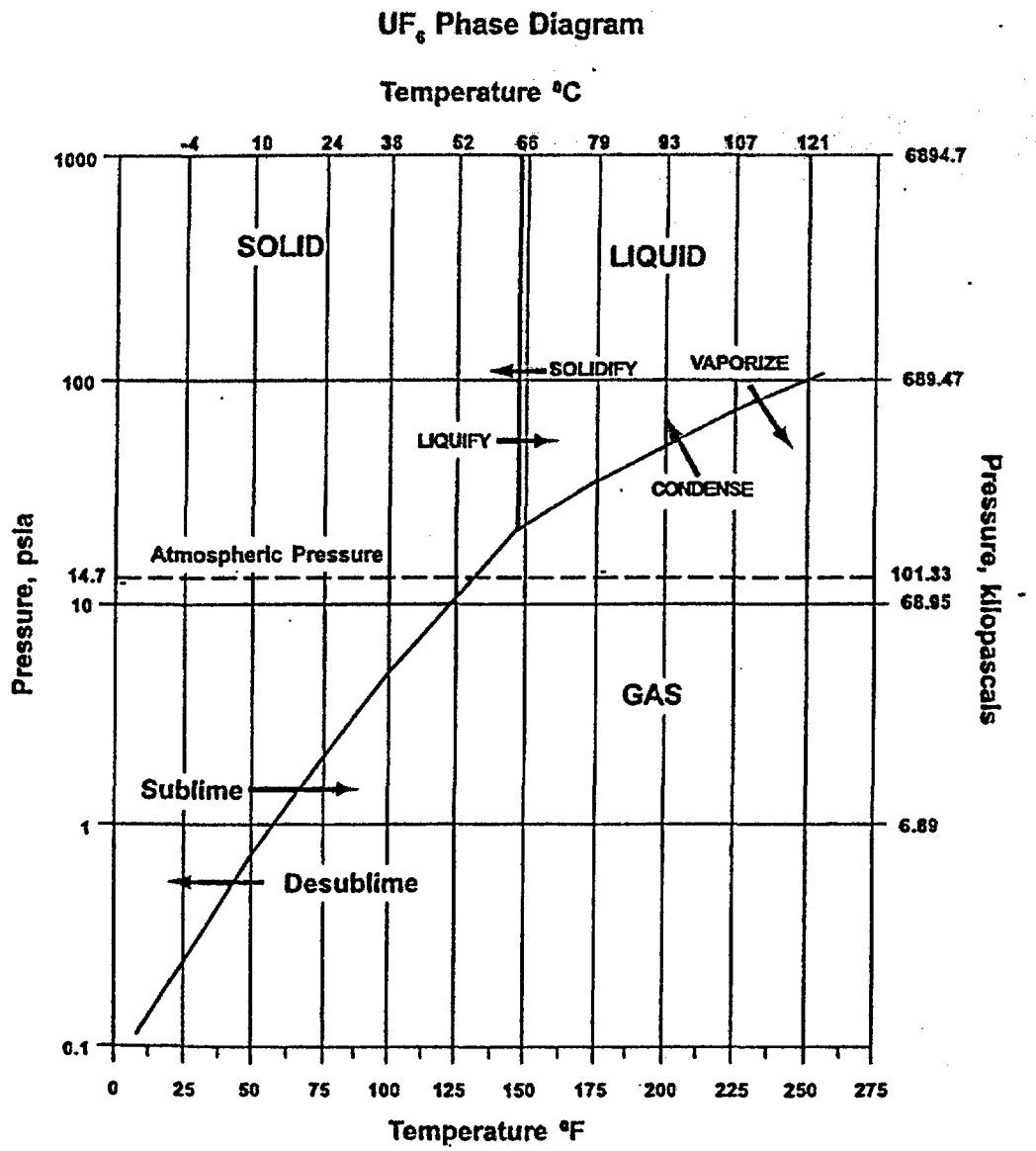

FIGURE 6.1-1
UF_6 PHASE DIAGRAM

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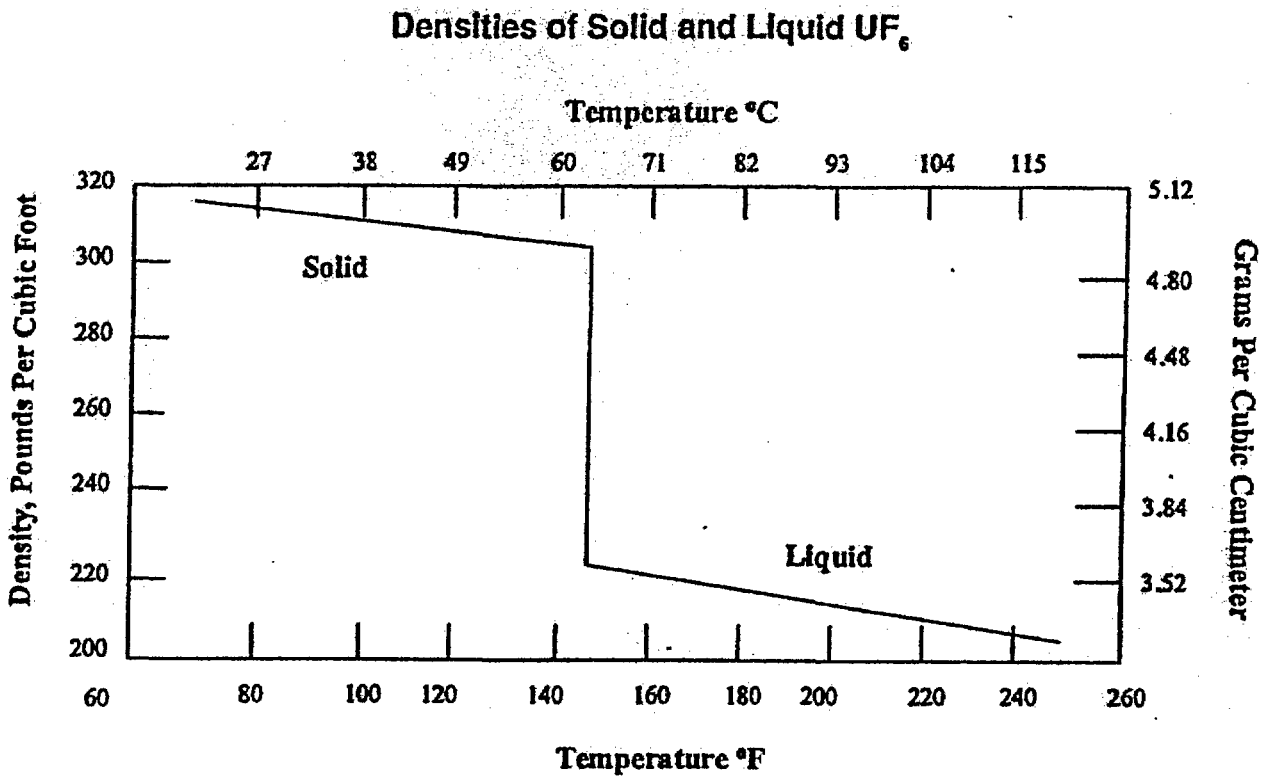

FIGURE 6.1-2
DENSITIES OF SOLID AND LIQUID UF_6

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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

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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). IROFS are identified in Chapter 3, Integrated Safety Analysis Summary.

7.1.1 Fire Protection IROFS

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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 Response Organization Qualifications, Drills and Training

The qualifications, drills and training of the fire brigade members who are part of the Emergency Response 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 response 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.

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7.3 FACILITY DESIGN

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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.

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7.5 FIRE PROTECTION AND EMERGENCY RESPONSE

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FIGURES

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