



OFFICE OF THE
GENERAL COUNSEL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 3, 2006

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Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Paul Abramson
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In the Matter of
LOUISIANA ENERGY SERVICES, L.P.
(National Enrichment Facility)
Docket No. 70-3103-ML

Dear Administrative Judges:

Attached please find a redacted copy of Staff Exhibit 50-M, "Louisiana Energy Services National Enrichment Facility Safety Evaluation Report Summary." Chapter 13 has been redacted in accordance with the Board's Order issued on March 27, 2006. Chapter 12 and 14, while summarizing proprietary information, did not contain any proprietary information and, therefore, do not require redaction prior to public disclosure. The Staff consulted with counsel for LES, who indicated that LES does not object to the redactions.

Sincerely,

A handwritten signature in black ink that reads "Margaret J. Bupp".

Margaret J. Bupp
Counsel for NRC Staff

Attachment: As stated

cc w/att: Mr. Rod Krich
James R. Curtiss, Esq.
Office of the Secretary
Office of Commission Appellate Adjudication

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**LOUISIANA ENERGY SERVICES
NATIONAL ENRICHMENT FACILITY
SAFETY EVALUATION REPORT SUMMARY**

SEPTEMBER 16, 2005

**LOUISIANA ENERGY SERVICES
NATIONAL ENRICHMENT FACILITY
DEVIATIONS FROM NRC GUIDANCE**

In Louisiana Energy Services' (LES') application for a license to construct and operate a uranium enrichment facility in Lea County, New Mexico, there were 3 areas identified that are deviations from U.S. Nuclear Regulatory Commission (NRC) guidance. Those areas involve an exemption request to fund the decommissioning funding plan incrementally and 2 areas related to the nuclear criticality safety (NCS) review.

Decommissioning Funding Plan Exemption Request

In Section 1.2.5 and 10.2.2 of the applicant's Safety Analysis Report (SAR), the applicant addressed an exemption request to 10 CFR 40.36 and 10 CFR 70.25 to provide incremental funding for decommissioning to reflect its phased approach for enrichment capacity at the facility and its expected depleted uranium tails generation rate. The applicant stated that it would initially provide funding for the projected cost of facility decontamination and decommissioning, assuming operation at full capacity, and disposition of the tails generated during the first three years of operation. Thereafter, the applicant will provide NRC with revised funding instruments for depleted uranium disposition on an annual forward-looking incremental basis. In the event that the applicant does not employ all projected modules as expected, updates required under 10 CFR 40.36 and 10 CFR 70.25 could reflect a corresponding reduction in the anticipated facility decommissioning costs based on the actual number of modules used. NRC staff will review revisions to the cost estimate and the financial instrument, which are presented in Section 10.2.2 of the SAR, before the applicant takes possession of licensed material. NRC staff will also review all subsequent revisions to the cost estimate and financial instruments.

Under 10 CFR 40.14 and 10 CFR 70.17, the Commission may grant exemptions from the requirements of the regulations as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. NRC staff evaluated the exemption request and determined that such exemption is not prohibited by law. Staff also determined that, because the incremental funding approach proposed by the applicant will provide funding for the all applicant's decommissioning obligations at any point time, the approach will not endanger life or property or the common defense and security. Because the incremental funding approach will reduce the applicant's expenses from having to fund a 30-year decommissioning obligation when, in actuality, the decommissioning obligations prior to the end of the 30-year operating period are less, the staff has determined that the proposed approach will be in the public interest by reducing unnecessary regulatory costs. Therefore, the staff will grant the requested exemption as provided in Section 1.2.5 of the SAR. A license condition will be included in the license that will address the applicant's commitments for updating the decommissioning funding plan over time. This license condition is discussed further in Section 10.3.1.10 of the Safety Evaluation Report (SER).

Deviation from Certain ANSI/ANS Series-8 NCS Standards Relating to Criticality Safety

As discussed in Section 5.3.2 of the SER, LES did not follow certain American National Standards Institute/American Nuclear Society Standards endorsed in NRC Regulatory Guide 3.71 "Nuclear Criticality Safety Standards for Fuel and Material Facilities." With regard to piping configurations containing aqueous solutions of fissile material, LES used a 1998 version of ANSI/ANS-8.1 instead of the 1995 version of ANSI/ANS-8.9, using validated methods to determine subcritical limits. In addition, the applicant used a newer version of the ANSI/ANS-8.1 standard (the 1998 version) rather than the version of the ANSI/ANS-8.1 standard (the 1988 version) that the NRC endorsed (with exception) in Regulatory Guide 3.71. NRC staff reviewed the differences between the two versions of ANSI/ANS-8.1 and found this approach acceptable. The applicant also committed to the following, concerning validation using ANSI/ANS-8.1-1998: "In addition, the details of validation should state computer codes used, operations, recipes for choosing code options (where applicable), cross-section sets, and any numerical parameters necessary to describe the input."

The applicant also used a newer version of ANSI/ANS-8.7 (1998 version) instead of the version of the standard endorsed by NRC in Regulatory Guide 3.7.1. NRC staff reviewed the differences between the two versions of ANSI/ANS-8.7 and determined that it was acceptable for the applicant to use the newer version without exception.

Based on the review of the information provided, the staff found the ANSI/ANS standards LES used instead of those identified in the applicable guidance were appropriate and consistent with a new, updated version of Regulatory Guide 3.7.1 currently being developed by the Staff.

Nuclear Criticality Safety Safe-By-Design Integrated Safety Assessment (ISA) Methodology

Staff evaluated the applicant's approach to ensuring that safe-by-design equipment provided an adequate safety margin in Section 3.3.3.2.2.2 of the SER. In Section 3.1.3.2 of the applicant's SAR, the applicant described a safe-by-design ISA method for selected equipment for NCS used to identify safe-by-design components, the failure of which would be highly unlikely. This approach was proposed as an alternative to identifying specific accident sequences as described in NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." The applicant described the connection between subcriticality and the safe-by-design ISA process for NCS. Using the safe-by-design ISA process, there are no accident sequences and, hence, items relied on for safety (IROFS) are not identified because it is highly unlikely these components would fail. Those safe-by-design components are considered items which may affect IROFS.

A qualitative determination of highly unlikely can apply to passive design component features of the facility that do not rely on human interface to perform the NCS function. Safe-by-design components are those components that by their physical size or arrangement have been shown to have an effective neutron multiplication factor (k_{eff}) less than 0.95. The definition of safe-by-design components encompasses two different categories of components. The first category includes those components that are safe-by-volume, safe-by-diameter, or safe-by-slab thickness (i.e., favorable geometry components). A set of generic, conservative NCS calculations has determined the maximum volume, diameter, or slab thickness that would result in a $k_{eff} < 0.95$. A favorable geometry component has a volume, diameter, or slab thickness that is less than the associated value for $k_{eff} < 0.95$. The components in the second category (i.e., non-favorable geometry components) require a more detailed NCS analysis to demonstrate k_{eff}

< 0.95. For the non-favorable geometry components, the design configuration is not bounded by the results of the generic, conservative NCS calculations for maximum volume, diameter, or slab thickness that would result in a $k_{eff} < 0.95$.

For failures of these passive safe-by-design components (i.e., both favorable geometry components and non-favorable geometry components) to be considered highly unlikely, those components must also meet the criterion that the only potential means to effect a change that might result in a failure to function would be to implement a design change (i.e., no potential failure mode exists). The evaluation of the potential to adversely impact the safety function of these design features includes consideration of potential mechanisms to cause bulging, corrosion, or breach of confinement/leakage and the subsequent accumulation of material. The evaluation further includes consideration of adequate controls to ensure that the double contingency principle is met. For each of these passive design components (i.e., both favorable geometry components and non-favorable geometry components), it must be concluded that there is no credible means to effect a geometry change that might result in a failure of the safety function and that significant margin exists.

For favorable geometry components, significant margin is defined as a margin of at least 10 percent, during both normal and upset conditions, between the actual design parameter value of the component and the value of the corresponding critical design attribute. For non-favorable geometry components, significant margin is defined as $k_{eff} < 0.95$, where $k_{eff} = k_{calc} + 3\sigma_{calc}$. This calculation of k_{eff} conservatively assumes the components are full of uranic breakdown material at maximum enrichment, with worst credible moderation, and with worst credible reflection.

These passive, safe-by-design features (i.e., both favorable geometry components and non-favorable geometry components) are considered items that may affect IROFS. As a result, Quality Level 1 requirements apply to these features. Also, the configuration management program required by 10 CFR 70.72 ensures the maintenance of the safety function of these features and assures compliance with both the double contingency principle and the defense-in-depth criterion of 10 CFR 70.64(b).

In Section 3.1.2 of the ISA Summary, the applicant provided a demonstration of meeting "highly unlikely" for NCS when using the safe-by-design ISA method to meet 10 CFR 70.65(b)(4). The demonstration of significant margin to meet "highly unlikely" was provided for each of the components listed in Tables 3.7-6 through 3.7-21 of the ISA Summary in the following classified documents: ETC4009554 through ETC4009559, ETC4009561, ETC4009565 through ETC4009567, ETC4009609, ETC4009614, ETC4009677, ETC4009679, ETC4009723, and ETC4009730. These classified documents are incorporated by reference into the ISA Summary. Also, the configuration management system required by 10 CFR 70.72, which is implemented by the facility Configuration Management Program, will ensure the maintenance of the safety function of these components and will assure compliance with both the double contingency principle and the defense-in-depth criterion of 10 CFR 70.64(b).

Staff reviewed classified information for all the applicant-identified safe-by-design components. For each piece of favorable geometry equipment, staff reviewed the dimensions provided to determine that it would meet the geometry criteria for significant margin. For each non-favorable geometry equipment, staff reviewed the appropriateness of the conservative assumption(s) and compared the calculated k_{eff} value versus the k_{eff} limit to determine that it would meet the criteria for significant margin. Therefore, NRC determined that the safe-by-

design components met the criteria for significant margin. The applicant slightly revised the classified information and then confirmed that all the information in the new classified documents met the criteria for using the safe-by-design ISA method for those components.

Based on the above review, the staff has reasonable assurance that: (1) the applicant used the safe-by-design ISA method appropriately; and (2) it is highly unlikely for an inadvertent criticality to occur with those safe-by-design components.

1.0 GENERAL INFORMATION

The purpose of this review is to determine whether the application includes: (1) an overview of the facility layout and a summary description of the proposed processes; (2) institutional information describing the identity of the applicant, its financial characteristics, the proposed activity, and foreign ownership, control, and influence; (3) provisions for obtaining liability insurance; (4) any special exemptions requested; (5) security of classified matter; and (6) a description of the site characteristics used in preparation of the emergency plan, Integrated Safety Analysis (ISA) Summary, and environmental report.

1.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The regulations in 10 CFR 30.33, 10 CFR 40.32, and 10 CFR 70.22 require each application for a license to include information on the proposed activity and the equipment and facilities that will be used by the applicant to protect health and minimize danger to life and property. In addition, the regulations in 10 CFR 70.65 require each application to include a general description of the facility, with emphasis on those areas that could affect safety, including identification of the controlled area boundaries.

The regulations in 10 CFR 30.32 and 10 CFR 40.31 require each application for a license to include institutional, financial qualifications, and foreign ownership, control, and influence information for the applicant. The regulations in 10 CFR Part 95 contain provisions for obtaining a facility security clearance. The regulations in 10 CFR 140.13b require applicants for uranium enrichment facilities to provide and maintain liability insurance.

The regulations in 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, and 10 CFR 70.65(b)(1) require each application to include a general description of the site, with emphasis on those factors that could affect safety (i.e., nearby facilities, meteorology, and seismology).

The acceptance criteria applicable to NRC's review of the facility and process description section of the application are contained in Sections 1.1.4.3, 1.2.4.3, and 1.3.4.3 of the "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520.

1.2 STAFF REVIEW AND KEY ISSUES

1.2.1 Review Process

Staff reviewed the information provided in Chapter 1 of the applicant's Safety Analysis Report (SAR) in accordance with guidance provided by NUREG-1520. The key information reviewed included descriptions of the facility and processes, institutional information, security of classified matter, liability insurance, and site description information. In performing the review, the staff prepared Requests for Additional Information (RAIs) (letter to Louisiana Energy Services (LES), dated April 19, 2004, and October 27, 2004), and resolved open items by reviewing the responses to the RAIs, dated May 10, 2004, and May 19, 2004, conducted an in-office review (memorandum from H. Graves to J. Gitter, dated June 29, 2004), and by followup meetings and conference calls with the applicant (memoranda from T. Johnson to J. Gitter, dated September 20, 2004, November 18, 2004, November 19, 2004, November 30, 2004, and

February 22, 2005). Based on the above communications with the staff, the applicant made appropriate revisions to the SAR and ISA Summary.

1.2.2 Key Areas of Review

Facility and Process Description

In Section 1.1.3 of the Safety Evaluation Report (SER), the staff evaluated the applicant's summary description of the proposed gas centrifuge uranium enrichment plant and processes. This description included discussion of the major chemical and mechanical processes to be used in the facility. The facility is proposing to use a gas centrifuge enrichment process based on European technology developed by Urenco Limited to enrich uranium from its natural isotopic concentration of about 0.7 percent uranium-235 (U-235) to 5 percent U-235. The proposed plant will have a nominal enrichment capacity of 3 million Separative Work Units (SWUs). (A SWU is a measure of the effort required to perform isotopic separation.) The process uses uranium in the chemical form of uranium hexafluoride (UF₆). Gaseous UF₆ enters a high-speed rotor at subatmospheric conditions where centrifugal forces press the heavier isotope of uranium, uranium-238 (U-238), to the outer wall of the rotor. The lighter isotope, U-235, remains closer to the center, away from the rotor wall. Internal scoops are used to collect the heavier and lighter fractions and circulate them to other centrifuges piped in a cascade arrangement. The staff concluded that the applicant has met the requirements and acceptance criteria applicable to this section.

Institutional Information

In Section 1.2.3 of the SER, the staff reviewed the applicant's corporate information including the identities of the general and limited partners and the financial qualifications of the applicant to construct and operate the facility to meet the U.S. Nuclear Regulatory Commission's (NRC's) safety requirements. The applicant is a Limited Partnership chartered in Delaware. The General Partners are Urenco Investments, Inc., a Delaware corporation and wholly owned subsidiary of Urenco Limited, a corporation owned by British, Dutch, and German interests, and Westinghouse Enrichment Company, LLC, a wholly owned subsidiary of Westinghouse Electric Company LLC. The Limited Partners include Westinghouse Enrichment Corporation and subsidiaries of Urenco, Entergy Corporation, Duke Energy Corporation, and Exelon Generation Company.

In Section 1.2.3.2 of the SER, staff evaluated foreign ownership, control, or influence (FOCI) with the assistance of the U.S. Department of Energy (DOE). In a letter from DOE, dated March 31, 2005, DOE recommended that the NRC waive the requirement for FOCI mitigation associated with the granting of a nuclear facility license to LES principally because there is an agreement between the United States (U.S.), United Kingdom, The Netherlands, and Germany to allow transfer of the Urenco technology into the U.S. The NRC staff accepted this finding by DOE based on an Interagency Agreement between NRC and DOE dated May 6, 2002.

In Section 1.2.3.3.1 of the SER, staff evaluated the applicant's estimates of the total cost of \$1.2 billion, in 2002 dollars, to construct the facility. Before starting the detailed review of the cost estimate, the staff conferred with the technical reviewers assigned to evaluate the support systems/structures necessary to support the safe operation of the facility to confirm that the

necessary systems had been identified in the SAR. The staff also conducted a detailed review of the Section 1.1.1 of the SAR, which provided a detailed description of each supporting structure/system, and then compared the support systems for each building with the systems identified in the cost estimate, to confirm that the cost estimate and the facility description were consistent. The staff concluded that the cost estimate is based on a reasonable estimate of the cost of the supporting systems and structures, as well as confirmed that all the major equipment necessary to support safe operation were included.

In Section 1.2.3.3.2 of the SER, the staff evaluated the applicant's financial qualifications for construction and operation of the proposed facility. The applicant made commitments that construction of the facility will not begin before funding is fully committed. Of this funding (equity and debt), the applicant will have in place, before construction, a minimum of equity contributions of 30 percent of the project's estimated costs of \$1.2 billion from the parents and affiliates of the partners, and firm commitments ensuring funds for the remaining project costs. The applicant plans to fund the construction phase of the project with a mix of approximately 50 percent debt and 50 percent equity contributions by the two major partners. The staff views the applicant's reliance on approximately 50 percent equity as a positive endorsement because, by contrast, some analogous construction projects rely on 100 percent financing, which often proves to be difficult to secure from financial institutions. The staff also reviewed income statements of the general and limited partners. The applicant has no reported income statements. The staff found that the partners have assets to support their respective equity ownership portions of LES. On December 3, 2003, the applicant announced that the first round of contracts with several U.S. nuclear power plants, including Exelon, were signed. These contracts represent at least 70 percent of the facility's first 10 years of production. The staff concluded that LES and its partner-owners appear to be financially qualified to build and operate the proposed facility, in accordance with 10 CFR 70.23(a)(5).

In Section 1.2.3.3.3 of the SER, the staff evaluated the applicant's approach for obtaining liability insurance to cover public claims arising from any occurrence, within the U.S. that causes, within or outside the U.S., bodily injury, sickness, disease, death, loss of, or damage to, property, or loss of use of property arising from the radioactive, toxic, explosive, or other hazardous properties of chemicals containing licensed material. The applicant proposed to have and maintain up to \$300 million to satisfy the 10 CFR 140.13b requirement. The applicant has already obtained a nuclear energy liability policy with a limit of \$1 million as a standby policy until the facility is ready to begin operations. At that time, the applicant will increase the amount to approximately \$300 million. Because full liability insurance coverage will not be provided until prior to receipt of licensed material, NRC staff will impose a license condition to ensure that proof of full liability insurance has been obtained prior to beginning operations.

In Sections 1.2.3.6 and 10.3.1.10 of the SER, staff evaluated an exemption request under 10 CFR 40.36 and 10 CFR 70.25 to provide incremental funding for decommissioning to reflect its phased approach for enrichment capacity at the facility and its expected depleted uranium tails generation rate. As discussed in Section 10.2.2 of the SAR (LES, 2005a), the applicant stated that it would initially provide funding for the projected cost of facility decontamination and decommissioning, assuming operation at full capacity, and disposition of the tails generated during the first three years of operation. Thereafter, the applicant will provide NRC with revised funding instruments for depleted uranium disposition on an annual forward-looking incremental basis. In the event that the applicant does not employ all projected modules as expected,

updates required under 10 CFR 40.36 and 10 CFR 70.25 could reflect a corresponding reduction in the anticipated facility decommissioning costs based on the actual number of modules used. Under 10 CFR 40.14 and 10 CFR 70.17, the Commission may grant exemptions from the requirements of the regulations as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. NRC staff evaluated the exemption request and determined that such exemption is not prohibited by law. Staff also determined that, because the incremental funding approach proposed by the applicant will provide funding for all the applicant's decommissioning obligations at any point in time, the approach will not endanger life or property, or the common defense and security. The staff will impose a license condition to address the applicant's commitments for updating the decommissioning funding plan over time. This license condition is discussed further in Section 10.3.1.10 of the SER.

In Section 1.2.3.7 of the SER, the staff reviewed and evaluated information provided by LES in the facility's proposed security procedures and controls to ensure that classified matter is used, processed, stored, reproduced, transmitted, transported, and destroyed in accordance with the requirements of 10 CFR Parts 25 and 95. The staff reviewed the applicant's Standard Practice and Procedures Plan (SPPP) for compliance with the requirements of 10 CFR Parts 25 and 95, by using "Standard Practice Procedures Plan Standard Format and Content for the Protection of Classified Matter for NRC Licensee, Certificate Holder and Others Regulated by the Commission." The staff concluded that the applicant provided sufficient information in its SPPP, and a facility clearance can be issued.

Site Description

In Section 1.3 of the SER, the staff reviewed information describing the proposed site. The site description review included information on the geographic, demographic, meteorological, geologic, hydrologic, and seismologic characteristics of the site and the surrounding area. The site description is a summary of the information that the applicant used in preparing the environmental report, emergency plan, and ISA Summary. The staff concluded that the applicant adequately described and summarized general site information and verified that the site description is consistent with the information used as a basis for the environmental report, emergency management plan, and ISA Summary.

2.0 ORGANIZATION AND ADMINISTRATION

The purpose of the review of the applicant's organization and administration is to ensure that the proposed management policies will provide reasonable assurance that the licensee plans, implements, and controls site activities in a manner that ensures the safety of workers, the public, and the environment. The review also ensures that the applicant has identified and provided adequate qualification descriptions for key management positions.

2.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, 10 CFR 70.23, and 10 CFR 70.62(d) require a management system and related administrative procedures for the effective implementation of health, safety, and environment (HS&E) protection functions concerning the applicant's corporate organization, qualifications of the staff, and adequacy of the proposed equipment, facilities, and procedures to provide adequate safety for workers, the public, and the environment. The acceptance criteria applicable to the U.S. Nuclear Regulatory Commission's (NRC's) review of the organization and administration section of the application are contained in Section 2.4.3 of the "Standard Review Plan for Fuel Cycle Facilities," NUREG-1520.

2.2 STAFF REVIEW AND KEY ISSUES

2.2.1 Review Process

Staff reviewed the information provided in Chapter 2 of the applicant's Safety Analysis Report (SAR) in accordance with guidance provided in NUREG-1520. The key information reviewed included organizational responsibilities and qualifications, management control, and transition from design and construction to operations. In preparing this review, staff prepared Requests for Additional Information (RAIs) (letter to Louisiana Energy Services, dated April 19, 2004) and resolved open items by reviewing the responses to the RAIs, dated May 19, 2004, February 17, 2005, and March 14, 2005. Based on the above communications, the applicant made appropriate revisions to the SAR.

2.2.2 Key Areas of Review

Organizational Responsibilities and Qualifications

In Section 2.3.1 of the Safety Evaluation Report (SER), the staff reviewed the applicant's functional description of specific organization groups responsible for managing the design, construction, and operation of the facility. The staff also reviewed the qualifications, responsibilities, and authorities for key supervisory and management personnel, along with a listing of the shift crew composition. Staff concluded that the applicant's organizational and administrative elements describe clear responsibilities and associated resources for the design, construction, and operation of the facility.

Management Control

In Section 2.3.2 of the SER, staff evaluated the applicant's administration procedures for effective implementation of health, safety, and engineering functions using written procedures,

and for reporting of unsafe conditions or activities. Staff also reviewed the applicant's written agreements with offsite emergency resources and its commitment to establish formal management measures to ensure availability of Items Relied on for Safety. Management control functions include a configuration management program for managing changes, a maintenance program, a personnel training program, a system of management audits and assessments, a Safety Review Committee that provides technical and administrative review of radiation and chemical safety programs, incident investigations, employee concerns, records management, and written agreements with offsite emergency response agencies. Staff concluded that the applicant has sufficient management controls to ensure that design, construction, and operation of the facility will meet the applicant's commitments for safety.

Transition from Design and Construction to Operations

In Section 2.3.3 of the SER, the staff evaluated the applicant's proposed changes to its organization to affect an orderly transition from design and construction activities to operations. Toward the end of construction, the focus of the organization will shift from design and construction to initial start-up and operation of the facility. As the facility nears completion, the applicant will staff the facility to ensure smooth transition from construction activities to operation activities. Urenco, the supplier of the gas centrifuge technology, will have personnel integrated into the organization to provide technical support during startup of the facility and transition into the operations phase. Staff concluded that the transition from design and construction to operations will be managed to ensure an orderly transfer of functions.

3.0 INTEGRATED SAFETY ANALYSIS (ISA) AND ISA SUMMARY

The purpose of this review is to ensure that the Integrated Safety Analysis (ISA) and ISA Summary meet the regulatory requirements specified in 10 CFR Part 70, Subpart H, "Additional Requirements for Certain Licensees Authorized to Possess a Critical Mass of Special Nuclear Material." The review determines whether appropriate hazards and baseline design criteria have been addressed; whether acceptable Items Relied on for Safety (IROFS), management measures, and likelihoods and consequences have been designated for higher-risk accident sequences; and whether, with IROFS, the performance requirements of 10 CFR 70.61 have been met. For those cases involving nuclear criticality safe-by-design components, the review determines whether the performance requirements of 10 CFR 70.61 are met through demonstration that failure of those components is highly unlikely. The review also determined whether programmatic commitments to maintain the ISA and ISA Summary are acceptable.

3.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The following regulatory requirements are applicable to the ISA and ISA Summary content:

1. 10 CFR 70.61 specified performance requirements for 10 CFR Part 70 applicants and licensees;
2. 10 CFR 70.62 requires an applicant to establish and maintain a safety program, and perform an ISA that demonstrates compliance with the performance requirements of 10 CFR 70.61;
2. 10 CFR 70.62(c) specifies requirements for conducting an ISA, including a demonstration that credible high-consequence and intermediate-consequence events meet the safety performance requirements of 10 CFR 70.61;
3. 10 CFR 70.64 specifies requirements for baseline design criteria and facility and system design and facility layout; and
4. 10 CFR 70.65(b) specifies the contents of an ISA Summary.

The acceptance criteria used during the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's ISA and ISA Summary are outlined in Sections 3.4.3.1 and 3.4.3.2 of NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," and NUREG-1513, "Integrated Safety Analysis Guidance Document."

3.2 STAFF REVIEW AND KEY ISSUES

3.2.1 Review Process

The staff reviewed the information provided in Chapter 3 of the Applicant's Safety Analysis Report (SAR) and the ISA Summary in accordance with the guidance provided by NUREG-1520. The key information reviewed included descriptions of the site, facility, and processes; hazards identification and analysis; accident sequences; IROFS and management measures; demonstration of meeting the 10 CFR 70.71 performance requirements; the ISA Summary

content; and Safety Program commitments. In performing the review, staff conducted an ISA in-office review (memorandum from T. Johnson to J. Giitter, dated March 24, 2004), conducted a site visit to the gas centrifuge uranium enrichment facility in Almelo, The Netherlands (memorandum from W. Troskoski to J. Giitter, dated April 7, 2004), prepared Requests for Additional Information (RAIs) (letter to the applicant dated April 19, 2004), and resolved open items by reviewing the responses to the RAIs dated May 19, 2005, and by followup conference calls with the applicant (memoranda from T. Johnson to J. Giitter, dated July 3, 2004, July 29, 2004, August 12, 2004, September 20, 2004, September 23, 2004, October 27, 2004, November 19, 2004, November 30, 2004, and February 22, 2005). Based on these communications with the staff, the applicant made appropriate revisions to the SAR and ISA Summary.

The staff also conducted detailed, vertical slice reviews of various accident scenarios, selected on a sampling basis, to confirm that the Safety Program and associated elements are adequately implemented by the applicant to achieve the performance requirements of 10 CFR 70.61. In accordance with the guidance in Section 3.5.2.3 of NUREG-1520, the vertical slice review examined how the ISA method was applied to a selected subset of facility processes in order to obtain reasonable assurance that ISA methods would be effective in the other processes not sampled by the staff.

Accident sequences related to chemical safety, nuclear criticality safety, and fire protection were selected for a detailed staff "vertical-slice" review based on gas centrifuge uranium enrichment process knowledge and professional judgement. The vertical-slice review examined how the ISA methods were applied and examined appropriate safety information not included in the ISA Summary. The vertical slice review included both high and intermediate consequence accident scenarios. The purpose of the review was to determine whether accident sequences, consequences, and likelihoods were reasonably determined, and whether appropriate IROFS and management measures were selected to limit the risk of the analyzed events (i.e., high-consequence events to "highly unlikely," and each intermediate-consequence events to "unlikely"). The results of the staff's vertical-slice review of a smart sample of accident sequences in each technical discipline will provide reasonable assurance that, if the methods described in the SAR, and discussed above, are appropriately applied by the applicant, all accident sequences and related IROFS will be identified by the applicant.

The staff also performed independent calculations of the consequences of a representative sample of accidents that are possible at the facility. These calculations were performed to ensure that the consequence analyses performed by the applicant were reasonable. The accident analyses are described in Appendix A of the Safety Evaluation Report (SER).

3.2.2 Key Areas of Review

Descriptions of the Site, Facility, and Processes

In Sections 3.3.1.1, 3.3.1.2, and 3.3.1.3 of the Safety Evaluation Report (SER), the staff reviewed the applicant's descriptions of the site characteristics, the facility design, and the processes to be used. The staff concluded that sufficient site, facility, and process information was provided by the applicant in the SAR and ISA Summary to use in determining appropriate

criteria for the safe design of principal structures, systems, and equipment, and for identifying hazards and developing appropriate accident sequences for use in the ISA.

Hazards Identification and Analysis

In Sections 3.3.3.2.2 and 3.3.3.3.1 of the SER, the staff evaluated the applicant's hazard identification process and analysis. The applicant's ISA uses the HAZOP method for identifying the hazards for UF₆ process systems, the Technical Services Building systems, the Centrifuge Assembly Building systems, and the Uranium Byproduct Storage Pad. The staff reviewed the applicant's HAZOP methodology and confirmed that it met the guidance in NUREG-1513 and generally acceptable industry practices. The HAZOP technique identifies and evaluates safety hazards in process plants and the technique requires detailed information concerning the design and operation of a process, and is typically used, as in this case, during or after the detailed design phase. Implementation of the technique involves the use of an interdisciplinary team and systematic approach to identify hazard and operability problems (i.e., accident sequences). The results of the HAZOP analysis are the team's findings, which include identification of the accident sequences and IROFS. As a result of the initial staff review, the applicant added a "safe-by-design" method to the ISA for application to passive design component features related to nuclear criticality safety. The staff subsequently determined that use of the HAZOP and "safe-by-design" ISA methods provided reasonable assurance that the applicant identified all accident sequences that could exceed the performance requirements of 10 CFR 70.61.

The applicant conducted two separate HAZOPs, one for classified systems and another for the non-classified portions of the process. The classified HAZOP was based the one performed for the operating European plants and was re-validated for the applicant's proposed facility.

The HAZOP analysis was applied to discrete process components. Radiological hazards identification considered the characteristics of uranium enriched to 5 percent (see Chapter 4 of this SER). Criticality hazards identification was performed for areas of the facility where fissile material is expected to be present (see Chapter 5 of this SER). Chemical hazards identification included those from licensed material and chemicals produced from licensed material, including chemical interactions (see Chapter 6 of this SER). Fire hazards identification considered in-situ and transient combustible sources and the use of fire barriers (see Chapter 7 of this SER). External hazards were considered at the site and facility level (see Chapter 3 of the SER). The applicant identified and listed hazards that could result in accident sequences exceeding the performance requirements of 10 CFR 70.61 in ISA Summary Tables 3.7-1 through 3.7-4. The HAZOP method is an acceptable hazard identification method per the guidance provided in NUREG-1513.

In SAR Section 3.1.3.2, the applicant described a safe-by-design ISA method for selected equipment for nuclear criticality safety (NCS) used to identify safe-by-design components, the failure of which would be highly unlikely. The applicant described the connection between subcriticality and the safe-by-design ISA process for NCS. Using the safe-by-design ISA process, there are no accident sequences and, hence, IROFS are not identified because it is highly unlikely these components would fail. Those safe-by-design components are considered items which may affect IROFS. As a result, Quality Level 1 requirements apply to these features. Also, the configuration management program required by 10 CFR 70.72 ensures the

maintenance of the safety function of these features and assures compliance with both the double contingency principle and the defense-in-depth criterion of 10 CFR 70.64(b).

Based on the above, the staff has reasonable assurance that the applicant used an acceptable hazard identification technique and identified all of the radiological hazards relating to possessing or processing licensed material; the chemical hazards of licensed material and hazardous chemicals produced from licensed material; facility hazards that could affect the safety of licensed material, and thus, present an increased radiological risk; and hazards related to process deviations or other events internal to the facility and credible external events, including natural phenomena.

Accident Sequences

In Sections 3.3.3.2.2 and 3.3.3.3.2 of the SER, the staff evaluated the applicant's accident sequences. In ISA Summary Tables 3.7-1 through 3.7-4, the applicant lists and describes the identified accident sequences and external and fire events for which the consequences could exceed the 10 CFR 70.61 performance requirements. ISA Summary Table 3.8-1 identifies each IROFS listed in the ISA Summary and how it protects against each accident sequence. The information is sufficient to determine how each accident sequence that could exceed the performance requirements of 10 CFR 70.61 is protected against by IROFS. The staff performed a review of selected high-consequence and intermediate-consequence events in the areas of chemical safety, criticality safety, and fire protection to confirm that the applicant had properly identified and analyzed accident sequences and the related consequences. The staff identified one unanalyzed accident sequence with consequences that exceeded the performance requirements that were overlooked by applicant. The accident sequence involves either a feed or depleted uranium cylinder shipment truck fire, as discussed in SER Section 7.3.2.2. The applicant addressed the accident sequence and identified additional IROFS.

The staff also reviewed the consequence analysis methods used to determine whether an accident sequence consequence exceeded a 10 CFR 70.61 performance requirement, the quantitative standards for chemical consequences, the assigned consequence categories and likelihood determinations used to determine the overall risk. The staff's review and independent evaluation of selected scenarios indicated that the applicant adequately applied appropriate consequence analysis methods.

IROFS and Management Measures

In Section 3.3.3.3.3 of the SER, the staff reviewed the applicant's IROFS and management measures. The only accident sequence types that can potentially result in intermediate or high consequences at this facility are loss of confinement events (i.e., caused by process upsets, human error, natural phenomena, fires, and external events) and criticality accidents (i.e., which are assumed to have a high consequence to the worker). Management measures are identified to ensure the IROFS are available and reliable when needed to perform their safety function (see SER Chapter 11). All IROFS are designated Quality Level 1 items under the applicant's approved Quality Assurance Program Description. Safe-by-design components do not have IROFS identified because a failure is considered highly unlikely (i.e., there is no credible way to effect a geometry change that might result in a failure of the safety function). However, safe-by-design components are considered items which may affect IROFS, and the configuration

management system required by 10 CFR 70.72, which is implemented by the facility Configuration Management Program, ensures the maintenance of the safety function of these features. The staff identified the need for a license condition to require the formal definition of IROFS boundaries.

Demonstration of Meeting the 10 CFR 70.61 Performance Requirements

In Section 3.3.3.3.4 of the SER, the staff reviewed the applicant's demonstration of meeting the performance requirements in 10 CFR 70.61. Based on the above information regarding hazards, accident sequences, IROFS, and management measures, the staff concludes that the applicant has: (1) identified hazards related to this type of facility; (2) identified credible events that could exceed the performance requirements of 10 CFR 70.61 through the application of appropriate accident identification method in accordance with NUREG-1513 (see SER Chapter 3); (3) identified appropriate chemical dose and radiological dose values for determining intermediate consequence and high consequence events (see SER Section 3.3.3.b.2); (4) determined the consequences in accordance with the fuel facility accident analysis guidance in NUREG-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," (see SER Chapters 3 and 6); (5) established appropriate definitions for likelihood; and (6) applied those definitions in an acceptable manner to demonstrate that intermediate consequence events are unlikely and high consequence events are highly unlikely. For safe-by-design components, the staff concludes that the applicant has identified hazards and demonstrated that failure of safe-by-design components will be highly unlikely.

ISA Summary Content

In Section 3.3.3.2.1 of the SER, the staff reviewed the content of the ISA to ensure that it met the requirements of 10 CFR 70.65(b). Staff review of the original application found that the ISA Summary was incorporated as part of the license application. As a result of staff comments, the applicant separated the ISA Summary information from the License Application and associated Safety Analysis Report. The ISA Summary now contains:

- A general description of the site with emphasis on those factors that could affect safety. This information is used to determine the external and natural phenomena loads that could be placed upon the facility.
- A general description of the facility with emphasis on those areas that could affect safety. This information describes the basic design basis for the facility and its ability to withstand those external and natural phenomena loads.
- A description of each process analyzed in the ISA in sufficient detail to understand the theory of operation and, for each process, the hazards identified in the ISA and a general description of the types of accident sequences.
- Information that demonstrates compliance with the performance requirements of 10 CFR 70.61, including a description of the management measures, requirements for criticality monitoring and alarms and the information regarding the baseline design criteria and defense-in-depth practices set forth in 10 CFR 70.64.

- A description of the team, qualifications, and the methods used to perform the ISA.
- A list briefly describing each IROFS in sufficient detail to understand their functions in relation to the performance requirements of 10 CFR 70.61.
- A description of the proposed quantitative standards used to assess consequences to an individual from acute chemical exposure to licensed material or chemicals produced from licensed material.
- A descriptive list that identifies all IROFS that are the sole item preventing or mitigating an accident sequence that exceed the performance requirements of 10 CFR 70.61.
- A description of the definitions of unlikely, highly unlikely, and credible, as used in the evaluations in the ISA.

The staff determined that the above information had been adequately described in the revised ISA Summary.

Safety Program Commitments

In Section 3.3.3 of the SER, the staff reviewed the applicant's proposed safety program commitments identified in SAR Section 3.0 to determine that the three elements of process safety information, integrated safety assessment, and management measures demonstrates compliance with the performance requirements of 10 CFR 70.61, that records are established and maintained to demonstrate compliance with 10 CFR 70.62(b) through (d), and that records are established and maintained documenting each discovery that an IROFS or management measure has failed to perform on demand or has degraded such that the performance requirements of 10 CFR 70.61 are not satisfied.

4.0 RADIATION PROTECTION

The purpose of this review is to determine whether the applicant's Radiation Protection (RP) program is adequate to protect the radiological health and safety of workers and to comply with the associated regulatory requirements in 10 CFR Parts 19, 20, 30, 40, and 70.

4.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The regulations applicable to performing a RP program review are the general and additional contents of application, as required by 10 CFR 19.12, 10 CFR Part 20, 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, 10 CFR 70.61, and 10 CFR 70.64. The acceptance criteria for the Nuclear Regulatory Commission's (NRC) review of the RP program are outlined in Sections 4.4.1.3, 4.4.2.3, 4.4.3.3, 4.4.4.3, 4.4.5.3, 4.4.6.3, 4.4.7.3, and 4.4.8.3 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520.

4.2 STAFF REVIEW AND KEY ISSUES

4.2.1 Review Process

Staff reviewed the information provided in Chapter 4 of the applicant's Safety Analysis Report (SAR) and the Integrated Safety Analysis (ISA) Summary in accordance with guidance provided by NUREG-1520. The key information reviewed included radiation protection program implementation; the As Low As Reasonably Achievable (ALARA) program; organization and personnel qualifications; written procedures; training; ventilation and respiratory protection programs; radiation survey and monitoring programs; and additional program requirements. In performing this review, the staff conducted an ISA in-office review (memorandum from T. Johnson to J. Giitter, dated March 24, 2004), prepared a Request for Additional Information (RAIs) (letter to LES dated April 19, 2004), and resolved open items by reviewing the responses to the RAIs, dated May 19, 2004, and by followup conference calls with the applicant (memorandum from T. Johnson to J. Giitter, dated September 23, 2004). Based on the above communications with the staff, the applicant made appropriate revisions to the SAR.

4.2.2 Key Areas of Review

Radiation Protection Program Implementation

Staff reviewed the radiation protection program implementation in Section 4.3.1 of the Safety Evaluation Report (SER). In Section 4.1 of the LES SAR, the applicant describes the proposed RP program for the proposed facility. The RP program is developed, documented, and will be implemented commensurate with the risk posed by a uranium enrichment operation that will meet the requirements of 10 CFR Part 20, Subpart B. The applicant will ensure that the RP program will remain independent of the facility's routine operations, and that it maintains its objectivity and is focused only on implementing sound RP principles necessary to achieve ALARA goals. The applicant will review the content and implementation of the RP program at least annually, in accordance with 10 CFR 20.1101(c). The RP program's organizational structure and the responsibilities of key program personnel are outlined in Section 4.1.1 of the SAR. The Plant Manager will be responsible for the protection of all persons against radiation

exposure resulting from facility operations and material, and for compliance with applicable NRC regulations and the facility license. The RP Manager will be responsible for implementing the RP program. The staff determined that the applicant has established and will maintain the RP program in accordance with the acceptance criteria in 10 CFR Part 20 and NUREG-1520.

ALARA Program

Staff reviewed the applicant's ALARA program implementation in Section 4.3.2 of the SER. The ALARA program will be implemented using written policies and procedures, to ensure occupational radiation exposures are maintained ALARA and that such exposures are consistent with the requirements of 10 CFR 20.1101. The goals of the ALARA program include maintaining occupational exposures, as well as environmental releases, as far below regulatory limits as is reasonably achievable. The applicant states that the RP Manager will be responsible for implementing the ALARA program and preparing an ALARA program evaluation report annually. This report will be submitted to the Plant Manager and the Safety Review Committee (SRC). The SRC will fulfill the duties of the ALARA Committee. The SRC will have at least five members, to include experts in operations, criticality safety, radiological safety, chemical safety, and industrial safety. The staff determined that the applicant established and will maintain an ALARA program in accordance with the acceptance criteria in 10 CFR Part 20 and NUREG-1520.

Organization and Personnel Qualifications

Staff reviewed the applicant's radiation protection organization and personnel qualifications in Section 4.3.3 of the SER. The applicant will employ only suitably trained RP personnel at the facility, by following the guidance in Regulatory Guides 8.2, "Guide for Administrative Practice in Radiation Monitoring," and 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable." Further information on personnel qualifications and training is provided in Sections 2.24, 2.3.3, and 11.3 of the SAR. The RP Manager will be responsible for establishing and implementing a personnel training program. The RP staff will be trained and qualified consistent with guidance provided in American National Standards Institute (ANSI)/American Nuclear Society (ANS) Standard 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants." The staff determined that the applicant will organize and staff an RP program in accordance with the acceptance criteria in 10 CFR Part 20 and NUREG-1520.

Written Procedures

In Section 4.3.4 of the SER, staff reviewed the applicant's use of written procedures for implementation of its RP program. Written procedures will be used for all operations involving licensed materials. The RP procedures will be reviewed and revised as necessary, to incorporate any facility or operational changes to the facility's ISA. The applicant will perform all work in Restricted Areas in accordance with a Radiation Work Permit (RWP). The applicant will also issue RWPs for activities involving licensed materials not covered by operating procedures, and where radioactivity levels are likely to exceed airborne radioactivity limits, or whenever deemed as necessary by the RP Manager. The staff concluded that the applicant will prepare written procedures and RWPs in accordance with the acceptance criteria in 10 CFR Part 20 and NUREG-1520.

Training

Staff reviewed the applicant's RP training program in Section 4.3.5 of the SER. The applicant has incorporated the provisions of 10 CFR 19.12 into the radiation training program, as outlined in Section 4.5.1 of the SAR. The requirements in 10 CFR 19.12 address required health physics information the applicant must make available to workers likely to receive exposures greater than 1 mSv (100 mrem) per year. An RP training program is designed and implemented to provide training to all personnel and visitors who enter Restricted Areas or Controlled Areas, unless provided with trained escorts. Retraining is performed for personnel requiring unescorted access to Restricted Areas on an annual basis, and as necessary, to address changes in policies, procedures, requirements, and the facility ISA. The staff determined that the applicant will train its employees in RP in accordance with the acceptance criteria in 10 CFR Part 20 and NUREG-1520.

Ventilation and Respiratory Protection Programs

Staff reviewed the applicant's ventilation and respiratory protection programs in Section 4.3.6 of the SER. The design criteria are described in Sections 3.4.9 and 3.5.1 of the ISA Summary. Filters to be used in the systems include pre-filters for dust removal, high-efficiency particulate air filters for removal of uranyl fluoride (UO_2F_2) aerosols, and activated carbon filters for hydrogen fluoride (HF) removal. To meet the respiratory protection requirements in 10 CFR Part 20, Subpart H, the applicant will prepare written procedures for the selection, fitting, issuance, maintenance, testing, training of personnel, monitoring, and record-keeping for individual respiratory protection equipment, in accordance with 10 CFR 20.1703(c)(4). The staff determined that the applicant established ventilation and respiratory protection programs in accordance with the acceptance criteria, and satisfies the regulatory requirements of 10 CFR Part 20, Subpart H.

Radiation Survey and Monitoring Programs

Staff reviewed the applicant's radiation survey and monitoring programs in Section 4.3.7 of the SER. The applicant has a radiation survey and monitoring program using prepared written procedures that will include an outline of the program objectives, sampling procedures, data analysis methods, types of equipment and instrumentation to be used, frequency of measurements, record-keeping and reporting requirements, and actions to be taken when measurements exceed 10 CFR Part 20 occupational dose limits, or the administrative limits established by the applicant.

Thermoluminescent dosimeters, supplied by a vendor holding dosimetry accreditation from the National Voluntary Laboratory Accreditation Program, will be required to be worn by all personnel who enter Restricted Areas. All personnel wearing external dosimetry devices will be evaluated for internal exposures via direct bioassay, indirect bioassay, or an equivalent technique. These doses will be evaluated at least annually. The applicant will sum the internal and external exposure values in accordance with 10 CFR 20.1202, using procedures based on the guidance in Regulatory Guides 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," and 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses."

The applicant will perform air sampling consistent with the guidance provided in Regulatory Guide 8.25, "Air Sampling in the Workplace;" NUREG-1400, "Air Sampling in the Workplace;" and ANSI/Health Physics Society Standard 13.1. Airborne activity levels in the Restricted Areas of the facility will be continuously monitored, with permanent air monitors designed to detect alpha emitters. The staff determined that the applicant has a radiation survey program consistent with the guidance contained in Regulatory Guide 8.24, "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication," and has established radiation survey and monitoring programs in accordance with the acceptance criteria in NUREG-1520 and 10 CFR Part 20.

Additional Program Requirements

Staff reviewed the applicant's commitments to additional program requirements in Section 4.3.8 of the SER. The applicant established a program to maintain records of the RP program, radiation survey results, and results of corrective action program referrals, RWPs, and planned special exposures.

The applicant will report, to the NRC, any event that results in an occupational exposure to radiation exceeding the dose limits in Part 20, within the time specified in 10 CFR 20.2202, 10 CFR 30.50, 10 CFR 40.60, and 10 CFR 70.74. The applicant will prepare and submit, to the NRC, an annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b).

The staff reviewed the ISA Summary and performed an on-site review of the ISA, and agree with the applicant that, as stated in Section 3.6 of the ISA Summary, the hazards from radioactivity were evaluated in the ISA and found to be of low consequence.

5.0 NUCLEAR CRITICALITY SAFETY

The purpose of this review is to determine whether the applicant's nuclear criticality safety (NCS) program is adequate to support safe design, construction, and operation of the facility, as required by 10 CFR Part 70. In addition, the purpose of this review is to determine whether the Integrated Safety Analysis (ISA) and ISA Summary meet the regulatory requirements specified in 10 CFR Part 70, Subpart H, "Additional Requirements for Certain Licensees Authorized to Possess a Critical Mass of Special Nuclear Material," for NCS.

5.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The regulatory basis for the NCS review is governed by 10 CFR 70.22; 70.24; 70.52; 70.61; 70.62, 70.64; 70.65; 70.72; and Appendix A to Part 70. The regulatory basis for the NCS ISA review is governed by 10 CFR 70.62 and 10 CFR 70.65. The acceptance criteria the U.S. Nuclear Regulatory Commission (NRC) uses for reviews of NCS are outlined in Sections 5.4.3.1 through 5.4.3.4 of NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." This includes the commitment to use NRC NCS Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities," which modified the use of the American National Standards Institute/American Nuclear Society (ANSI/ANS) Series-8 NCS standards. The acceptance criteria the NRC uses for NCS reviews of the ISA program and ISA Summary are outlined in Sections 3.4.3.1 and 3.4.3.2 of NUREG-1520.

5.2 STAFF REVIEW AND KEY ISSUES

5.2.1 Review Process

Staff reviewed the information provided in Chapter 5 of the applicant's Safety Analysis Report (SAR) and the ISA Summary in accordance with guidance provided by NUREG-1520. The key information reviewed identified the management of the NCS program, NCS organization and administration, NCS management measures, and the NCS program. The specific areas of the NCS program were: NCS methodologies and technical practices, NCS criticality accident alarm system, NCS subcriticality of operations and margin of subcriticality for safety, NCS baseline design criteria, NCS in the ISA Program and the ISA Summary, and additional NCS program commitments.

In performing this review, staff attended the ISA approach meeting in Washington, DC (memorandum from T. Johnson to J. Giitter, dated March 3, 2004), conducted an NCS and ISA Summary in-office review in Marlboro, MA (memorandum from T. Johnson to J. Giitter, dated March 25, 2004), conducted an NCS and ISA Summary in-office review at the Urenco gas centrifuge uranium enrichment plant in Almelo, The Netherlands (memorandum from W. Troskoski to J. Giitter, dated April 7, 2004), prepared a Request for Additional Information (RAIs) (letter to LES dated April 19, 2004), and resolved open NCS and ISA Summary items by: (1) reviewing the responses to the RAIs (letters to NRC, dated May 7, 2004, May 19, 2004, September 30, 2004 (classified and unclassified), October 4, 2004, November 20, 2004 (classified and unclassified), December 10, 2004 (classified), January 18, 2005, February 28, 2005, March 28, 2005); (2) followup conference calls with the applicant (memoranda from T. Johnson to J. Giitter, dated April 22, 2004, July 20, 2004, July 29, 2004, September 20, 2004,

November 22, 2004, and April 11, 2005); (3) held a meeting in Rockville, MD (memorandum from T. Johnson to J. Giitter, dated July 21, 2004); and (4) conducted in-office reviews in Washington, DC (memoranda from T. Johnson to J. Giitter, dated September 15, 2004, February 3, 2005, and March 1, 2005). Based on the above communications with the staff, the applicant made appropriate revisions to the Safety Analysis Report and ISA Summary.

5.2.2 Key Areas of Review

Management of the NCS Program

Staff reviewed the applicant's proposed management of the NCS program in Section 5.3.3 of the Safety Evaluation Report (SER). Staff reviewed the applicant's proposals to develop, implement, and maintain an NCS program that would prevent inadvertent criticalities, maintain criticality limits and safety parameters, prepare appropriate NCS analyses and evaluations, ensure nuclear processes are subcritical and maintain approved margins of safety, comply with the performance requirements of 10 CFR 70.61, establish and maintain Items Relied on for Safety (IROFS), conduct NCS training, and comply with the NCS baseline design criteria in 10 CFR 70.65(b). Based on its review of the information provided, staff found that the applicant committed to adequate management of the NCS program.

NCS Organization and Administration

Staff reviewed NCS organization and administration in Section 5.3.4 of the SER. Staff reviewed the applicant's proposed organization during and after the design phase of the facility. Staff reviewed the positions, responsibilities, and qualification requirements related to NCS for the facility. Based on its review of the information provided, staff concluded that the applicant described an adequate NCS organization and associated administration to meet the requirements of 10 CFR 70.22(a)(6).

NCS Management Measures

Staff reviewed NCS management measures in Section 5.3.5 and Chapter 11 of the SER. Staff reviewed the applicant's proposed management measures (i.e., configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements). These management measures are intended to ensure that IROFS will be reliable and available to perform their intended function when needed. Based on its review of the information provided, staff concluded that the applicant described adequate NCS management measures to meet the requirements of 10 CFR 70.62(d).

NCS Program

The applicant's NCS program included the commitments to prevent an inadvertent criticality and to respond to an inadvertent criticality and descriptions of how to meet those commitments. The applicant will limit production at the facility up to 5 weight (wt) percent enriched uranium.

- Methods and Technical Practices

Staff reviewed the NCS methods and technical practices in Sections 5.3.6.1 of the SER. The applicant's proposed methods and technical practices include: application of the double contingency principle, as stated in ANSI/ANS 8.18, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors;" establishment of NCS limits; preparation of NCS calculations prepared using a validated methodology; and establishment of safety parameters: (i.e., enrichment, geometry and volume, moderation, mass, reflection, and interactions with other components that potentially may contain fissionable material).

In Table 5.3-1 of the SER, staff reviewed the applicant's proposed critical and safe values for safety parameters based on 6.0 wt percent enrichment (i.e. these are conservative values because production will be limited to 5 wt percent) against the values in ANSI/ANS 8.1 at 5.0 and 10.0 wt percent enrichment. The staff determined that the applicant's values in Table 5.3-1 of the SER were consistent with the values in ANSI/ANS 8.1.

Based on its review of the information provided, staff found the applicant's NCS methods and technical practices to be acceptable.

- NCS Criticality Accident Alarm System

Staff reviewed NCS criticality accident alarm system (CAAS) in Section 5.3.6.2 of the SER. The Technical Services Building, three Cascade Halls, Cylinder Receipt and Dispatch Building, Blending and Liquid Sampling Area, and UF₆ Handling Area are the only areas in the facility where an inadvertent criticality could occur. A CAAS will be provided to detect and alarm if an inadvertent criticality occurs in an area where uranium at or above the 10 CFR 70.24 mass limits will be handled, used, or stored. The CAAS will be designed, installed, and maintained in accordance with ANSI/ANS 8.3, "Criticality Accident Alarm System," as modified by Regulatory Guide 3.71. Based on its review of the information provided, staff concluded that the applicant described an adequate NCS CAAS to meet the requirements in 10 CFR 70.24 and 70.65(b)(4).

- NCS Subcriticality of Operations and Margin of Subcriticality for Safety

Staff reviewed NCS subcriticality of operations and margin of subcriticality for safety in Section 5.3.6.3 of the SER. Staff reviewed the applicant's use of the MONK8A Monte Carlo code for its NCS analyses. The applicant conducted a code validation, but also committed to submit, by December 30, 2005, a revision to the code validation that will meet ANSI/ANS 8.1, including details of validation that state computer codes used, operations, recipes for choosing code options (where applicable), cross-section sets, and any numerical parameters necessary to describe the input. The applicant will calculate the effective neutron multiplication factor, k_{eff} , using the equation: $k_{\text{eff}} = k_{\text{calc}} + 3\sigma_{\text{calc}} < 0.95$, where k_{calc} is the calculated neutron multiplication factor and σ_{calc} is the standard deviation of the calculation. Thus, the applicant will use a margin of subcriticality for safety of 0.05, when performing NCS calculations. This approach is based on the guidance in NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Computational Methodology." Based on its review of the information provided, staff concluded that the applicant adequately described how it assures subcriticality of operations under normal and credible abnormal conditions and defined an adequate margin of subcriticality for safety to meet the requirements of 10 CFR 70.61(d).

- NCS Baseline Design Criteria

Staff reviewed NCS baseline design criteria in Section 5.3.6.4 of the SER. Staff reviewed the methods used for criticality control in the NCS program. All process and storage systems will be designed and maintained with sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before an inadvertent criticality is possible (i.e., adherence to the double contingency principle). The major NCS controlling parameters used in the facility will be enrichment control, geometry control, moderation control, or limitations on the mass as a function of enrichment. Based on its review of the information provided, staff concluded that the applicant described criticality control, including adherence to the double contingency principle, to meet the NCS baseline design criteria of 10 CFR 70.64(a)(9).

- NCS in the ISA Program and in the ISA Summary

Staff reviewed NCS in the ISA program and in the ISA Summary in Section 5.3.6.5 of the SER. Staff reviewed the ISA method that the applicant used for NCS when performing the ISA of the process accident sequences. In this ISA method, the applicant identified the hazard, developed a risk matrix with consequence and likelihoods, defined "highly unlikely" and developed IROFS and general management measures to make NCS accident sequences meet "highly unlikely." The consequences of an inadvertent criticality were conservatively assumed to be high for the workers. For accident sequences postulated to result in an inadvertent criticality, IROFS were specified to ensure subcriticality under all normal and credible abnormal conditions and general management measures were specified. Staff reviewed the ISA method that the applicant used for NCS when performing the ISA for safe-by-design components. In this ISA method, "highly unlikely" is achieved with a significant margin and other conditions (i.e., there is no credible way to change the applicable geometric parameters without effecting a design change), rather than with accident sequences, IROFS, and management measures.

In Section 5.3.6.5.1.1 of the SER, staff reviewed the applicant's ISA Summary, which included credible NCS accident sequences and applicable IROFS to meet 10 CFR 70.61. Staff sorted the applicant's NCS accident sequences into groups of similar sequences. Staff evaluated each group of sequences regarding assumptions, descriptions, IROFS, and index scores. For those groups with sole NCS IROFS, staff determined that it was acceptable because: (1) failure of the IROFS would have to occur many times before a criticality could occur; (2) there is a limited number of cranes that can be used to move product cylinders; (3) the IROFS does not allow the operator to make qualitative judgements about the amount of moderation present; (4) a piece of equipment that meets the safe-by-design ISA method for NCS for moving/storing a container exists; or (5) more than one operator would have to independently incorrectly set the enrichment controls. Based on its review, staff found reasonable assurance that: (1) the applicant identified all hazards and accident sequences for NCS; and (2) the applicant's IROFS and management measures will ensure that it is highly unlikely for each accident sequence to occur (i.e., inadvertent criticality).

Staff reviewed the applicant's ISA Summary, which included a demonstration of meeting "highly unlikely" for NCS when using the safe-by-design ISA method to meet 10 CFR 70.65(b)(4). The demonstration of significant margin to meet "highly unlikely" was provided for each of the components listed in Tables 3.7-6 through 3.7-21 of the ISA Summary in classified documents,

which were incorporated by reference into the ISA Summary. Staff reviewed the classified information for all the applicant-identified safe-by-design components. For favorable geometry equipment, staff reviewed the dimensions provided to determine that it would meet the geometry criteria for significant margin. For non-favorable geometry equipment, staff reviewed the appropriateness of the conservative assumption(s) and compared the calculated k_{eff} value versus the k_{eff} limit criteria to determine that it would meet the criteria for significant margin. Therefore, staff determined that the safe-by-design components met the criteria for significant margin. Based its review, staff has reasonable assurance that: (1) the applicant used the safe-by-design ISA method appropriately; and (2) it is highly unlikely for an inadvertent criticality to occur with those safe-by-design components.

- Additional NCS Program Commitments

Staff reviewed additional NCS program commitments in Section 5.3.6.6 of the SER. Staff reviewed the applicant's additional commitments including reporting criteria for NCS deficiencies involving the loss or unavailability of IROFS, the loss of meeting the double contingency principle, evaluating changes to the NCS program, and record retention. Based on its review of the information provided, staff found the applicant's information regarding additional NCS program commitments to be acceptable and the staff concluded that the applicant described reporting criteria to meet the requirements of Appendix A to 10 CFR Part 70.

6.0 CHEMICAL PROCESS SAFETY

The primary purpose of this review is to determine that the applicant has designed a facility that will adequately protect workers, the public, and the environment during normal operations, and also against chemical hazards of licensed material and its byproducts. The applicant must also protect against facility conditions or operator actions that can affect the safety of licensed materials and thus present an increased chemical risk.

6.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The regulatory bases for the contents of the application are 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, and 70.65. In addition, the chemical-process safety review should provide a determination of compliance with 10 CFR 70.61, 70.62, and 70.64. The acceptance criteria for the U.S. Nuclear Regulatory Commission's (NRC's) review of chemical-process safety for the proposed facility are outlined in Section 6.4.3 of NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility."

6.2 STAFF REVIEW AND KEY ISSUES

6.2.1 Review Process

The NRC staff reviewed the information provided in Chapter 6 of the applicant's Safety Analysis Report (SAR) and the Integrated Safety Analysis (ISA) Summary in accordance with the guidance provided in NUREG-1520. The key information reviewed included the process description, chemical accident sequences, chemical accident consequences, items relied on for safety (IROFS), management measures, and baseline design criteria. The staff conducted an ISA in-office review to have a better understanding of the process and safety requirements (memorandum from T. Johnson to J. Giitter, dated March 24, 2004), prepared Requests for Additional Information (letter to LES dated April 19, 2004), resolved open items by reviewing the responses to the RAIs dated May 19, 2004, and by a followup conference call with the applicant (memoranda from T. Johnson to J. Giitter, dated November 30, 2004). Based on the above communications with the staff, the applicant made appropriate revisions to the SAR and ISA Summary.

6.2.2 Key Areas of Review

Process Description

In Section 6.3.1 of the Safety Evaluation Report (SER), the staff reviewed the applicant's plant process to enrich natural uranium hexafluoride (UF_6) by separating a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream enriched in the uranium-235 (U^{235}) isotope and a tails stream depleted in the U^{235} isotope. The process, entirely physical in nature, is a mechanical separation of isotopes using a fast rotating cylinder (centrifuge) based on a difference in centrifugal forces from differences in molecular weight of the uranic isotopes. No chemical changes nor nuclear reactions take place. The feed, product, and tails streams are all in the form of UF_6 .

UF₆ is delivered to the plant in American Nuclear Standards Institute (ANSI) N14.1, Standard for Nuclear Materials - Uranium Hexafluoride - Packaging for Transportation," Type 48X or 48Y cylinders. The enrichment process proposed by the applicant, housed in the Separation Building, is comprised of four major systems: a UF₆ Feed System, a Cascade System, a Product Take-off System, and a Tails Take-off System. Other product-related functions include the Product Liquid Sampling System and Product Blending System. Supporting functions include sample analysis, equipment decontamination and equipment rebuild, liquid effluent treatment, and solid waste management. The product is shipped in ANSI N14.1, Type 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 UF₆ or hydrogen fluoride (HF) are minimized, because leakage would typically be inward to the system. During sampling operations, UF₆ is liquified within an autoclave that provides the heating required to homogenize the material for sampling. The autoclave is an American Society of Mechanical Engineers (ASME), Section VIII, Division 1, "Boiler and Pressure Vessel Code"-rated pressure vessel that serves as a secondary containment for the UF₆ product cylinder while the UF₆ is in liquid state.

The only chemical present in significant quantities in the plant is UF₆, and it constitutes the main hazard in this facility. Any UF₆ that is released to the environment will react exothermically with water vapor, present in air, producing uranyl fluoride (UO₂F₂) and HF. The staff concluded that the applicant provided sufficient information to describe the chemicals and process systems to be used at the facility.

Chemical Accident Sequences

In Section 6.3.2 of the SER, staff reviewed the applicant's chemical accident sequences. In ISA Summary Section 3.7, Table 3.7-1, identifies the chemical accident sequences, and Table 3.7.2 provides a narrative description of the accident sequences. The chemical accident sequences covered the Tails Take-Off, UF₆ Feed, Product Take-Off, Product Blending and Liquid Sampling, Ventilated Room, Chemical Laboratory, Cylinder Preparation Room, Contingency Dump, Cascade, and the Centrifuge Test/Centrifuge Post-Mortem areas. A total of 36 different chemical accident sequences were identified by the applicant. The accident sequences covered the range of events that could result in a loss of containment of UF₆ and the hazardous chemicals produced from UF₆ (i.e., hydrogen fluoride, uranyl fluoride, and interactions with organic materials such as hydrocarbons). The accident sequences addressed both intermediate- and high-consequence events. Staff review of the process and hazards involved did not identify any chemical accident sequences overlooked by the applicant. The staff concluded that the applicant identified appropriate chemical accident sequences based on the applicant's use of an approved process hazards analysis method (HAZOP) to identify those sequences and the results of the above staff review.

Chemical Accident Consequences

In Section 6.3.3 of the SER, the staff reviewed the applicant's description of the chemical accident consequences. The applicant proposed chemical exposure limits based on the soluble uranium values referenced in NUREG-1391, "Chemical Toxicity of Uranium Hexafluoride

Compared to Acute Affects of Radiation," and proposed HF chemical exposure limits based on Acute Exposure Guideline Level (AEGLE) values, in a manner consistent with NUREG-1520, Table A-5, "Consequence Severity Categories Based on 10 CFR 70.61." The applicant also proposed a worker-exposure strategy that incorporates 10-minute AEGLE values for HF, as used in NUREG-1391 (NRC, 1991). The staff found this approach to be acceptable for the determination of compliance with the performance criteria of 10 CFR 70.61.

The applicant used the methods prescribed in NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," to determine the source terms. Many source term values are in the classified portion of the ISA. Staff review of the ISA and supporting documentation found the source term values to be reasonable. Site boundary atmospheric dispersion factors were generated based on Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." The applicant also used modeling methods for source-term determination, release fractions, dispersion factors, and meteorological conditions, as prescribed in Regulatory Guide 1.145. The staff concluded that the applicant's proposed method for source-term determination was acceptable.

Items Relied on for Safety

In Section 6.3.4.1 of the SER, the staff reviewed the applicant's chemical process IROFS. ISA Summary Section 3.7, Table 3.7-2, describes each of the applicant-identified accident sequences and the specific IROFS that are applied to prevent or mitigate the consequences of those accident sequences. ISA Summary Table 3.8-1 describes the safety functions of all identified IROFS and the specific accident sequences that take credit for each IROFS. The staff reviewed the listed IROFS and the process descriptions and process flow diagrams provided in ISA Summary Sections 3.4 and 3.5 to identify where each IROFS would be used and how the IROFS would function to prevent or mitigate the consequences of the identified accident sequences. The identified IROFS provide protection to prevent a loss of confinement of licensed material during operation of the facility. Based on this system level review and the staff's on-site visit to a similar gas centrifuge uranium enrichment facility in Almelo, The Netherlands, which uses the same technology and process systems, the staff concludes that the applicant has identified chemical-process IROFS to prevent the consequences of accident sequences involving the chemical hazards of licensed material and hazardous chemicals produced from licensed material.

Management Measures

In Section 6.3.4.2 of the SER, the staff reviewed the applicant's management measures for its chemical process IROFS. The applicant identified management measures to ensure the availability and reliability of chemical-safety IROFS in SAR Section 6.4. The applicant's Quality Assurance (QA) Program is described in Appendix A of the SAR. The applicant states that it will meet the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the guidelines of ASME QA Standard NQA-1-1994, "Quality Assurance Program Requirements for Nuclear Facilities." The applicant further states that it will apply all sections of the QA Program to the chemical-safety IROFS. Chapter 11 of the SER provides more detail of the management measures to be used at the facility.

Baseline Design Criteria

In Section 6.3.5 of the SER, the staff reviewed the applicant's information related to chemical process IROFS baseline design criteria. In the SAR and ISA Summary, the applicant provided design basis information for chemical-process-safety IROFS identified for the proposed facility. The only chemical of concern is UF_6 . Details of design and safety features of all chemical process systems are found in Chapter 3, ISA Summary. The applicant's design of the chemical process systems includes numerous controls, in addition to the IROFS, for maintaining safe conditions during operation. Based on the need to operate at, and maintain, a significant vacuum throughout the gaseous portion of the process, and the limited inventories of licensed material contained in any portion of the gaseous process, the staff concluded that the design basis provides for adequate protection against chemical risk.

The staff reviewed the results of the applicant's HAZOP analysis as discussed in SER Chapter 3. This method is widely used in the chemical industry during the design phase to identify operability and safety issues and is identified as an acceptable method in NUREG-1513, "Integrated Safety Analysis Guidance Document." As applied to the gas centrifuge uranium enrichment process, the HAZOP considered a variety of internal process, facility and external hazards that could breach the process and release licensed material and hazards chemicals produced from licensed materials. The results of the applicant's ISA are presented in ISA Summary Table 3.7-1. The table contains information concerning the accident sequences identified as a result of the HAZOP, the unmitigated risk of each applicant identified accident sequence, and the IROFS applied to prevent or mitigate the accident sequence. The staff also reviewed selected high-consequence and intermediate consequence accident scenarios to confirm that chemical events that could exceed the performance requirements of 10 CFR 70.61 were addressed.

Based on the above, the staff concluded that the applicant's design provides for adequate protection against chemical risks produced from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material, and meets the requirements of 10 CFR 70.64(a)(5).

7.0 FIRE SAFETY

The purpose of this review is to determine, with reasonable assurance, whether the applicant has designed a facility that provides adequate protection against fires and explosions that could affect the safety of licensed materials and thus present an increased radiological risk. The review should also establish that the applicant has considered the radiological consequences of fires and will institute suitable safety controls to protect workers, the public, and the environment.

7.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The regulatory basis for the fire safety review is governed by 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, and 10 CFR 70.65. In addition, the staff's fire safety review focuses on elements of compliance with 10 CFR 70.61, 70.62, and 70.64. The acceptance criteria the U.S. Nuclear Regulatory Commission (NRC) uses for reviews of fire safety are outlined in Sections 7.4.3.1 through 7.4.3.5 of NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility."

7.2 STAFF REVIEW AND KEY ISSUES

7.2.1 Review Process

Staff reviewed the information provided in Chapter 7 of the applicant's Safety Analysis Report (SAR) and the "Integrated Safety Analysis (ISA) Summary" in accordance with guidance provided by NUREG-1520. The key information reviewed identified potential fire hazards, the accident scenarios that cause a fire to occur, identified Items Relied on for Safety (IROFS) to mitigate or prevent fires and subsequent radioactive or chemical releases, and the fire prevention and protection features to be used at the plant. In performing this review, the staff conducted an ISA in-office review (memorandum from T. Johnson to J. Giitter, dated March 24, 2004, pp. 4-5), prepared Requests for Additional Information (RAIs) (letter to Louisiana Energy Services, dated April 19, 2004), and resolved open items by reviewing the responses to the RAIs, dated May 19, 2004, and by followup conference calls with the applicant (memoranda from T. Johnson to J. Giitter, dated July 13, 2004, and September 10, 2004). Staff also reviewed the applicant's baseline needs assessment of the response to fire and related emergencies, dated May 28, 2004, and prepared a calculation verifying applicant information on the effects of diesel fuel fires around uranium hexafluoride cylinders (memorandum from R. Wescolt to file dated March 22, 2005). Based on the above communications with the staff, the applicant made appropriate revisions to the SAR and ISA Summary.

7.2.2 Key Areas of Review

Process Fire and Special Hazards

Staff evaluated the facility's potential process fire and special hazards in Section 7.3.1 of the Safety Evaluation Report (SER). These hazards include the following:

UF₆: UF₆ is not flammable and does not disassociate to flammable constituents under conditions at which it will be handled at the facility. UF₆ does not react with oxygen, nitrogen,

carbon dioxide, or dry air, but does react with water or water vapor. Hydrocarbons can be explosively oxidized if they are mixed with UF_6 in the liquid state or at elevated temperatures. These interactions would be minimized because liquid UF_6 is present only in the sampling stations not elsewhere in the plant. For this reason, non-fluorinated hydrocarbon lubricants are not utilized in the UF_6 processes at the facility. UF_6 pumps are lubricated using a perfluorinated polyether oil that is referred to by the manufacturer's trade name, Fomblin oil.

Hydrogen Fluoride (HF): HF is a byproduct of the chemical reaction of UF_6 with water vapor. HF is extremely reactive in both gaseous and aqueous form. HF by itself is not flammable nor combustible. It can, however, react exothermically with water to generate sufficient heat to ignite nearby combustibles.

Uranyl Fluoride (UO_2F_2): UO_2F_2 is also a byproduct of the chemical reaction of UF_6 with water vapor. UO_2F_2 is stable in air to 300° C (572° F). It is not flammable nor combustible and will not decompose to combustible constituents under conditions that will exist at the facility.

Centrifuge Machines and Components: The only combustibles of any significance are the electrical cabling going to the drive motors. Therefore, any fire originating in one of the cascades will most likely result in limited damage to the centrifuge and its components, resulting in a small release.

Control Room: The control room will be provided with automatic smoke detection throughout. Additionally, the control room will house the fire alarm control panel and will be continuously staffed. Hand portable fire extinguishers will also be provided in the Control Room. IROFS boundaries will include appropriate electrical separation from normal instrument and control functions to ensure that fire-induced spurious actuations do not occur. Based on the current design, all active engineered components that are IROFS will fail in the safe configuration.

Storage and Handling of UF_6 : UF_6 cylinders are stored or handled in the Uranium Byproduct Cylinder (UBC) storage pad; the Cylinder Receipt and Dispatch Building (CRDB); the UF_6 handling areas; and the blending and liquid-sampling areas. On the UBC storage pad, fire concerns include the cylinder transport vehicle, a fire exposure from nearby vegetation, and fire exposure from a nearby vehicle accident. In the CRDB, the primary fire concern was from a truck fire at the loading dock. The applicant analyzed this and determined that unprotected cylinders could be adequately protected by storing them at least 1 meter (m) [3.3 feet (ft)] from the edge of the loading dock, and protecting cylinders on trucks with U.S. Department of Transportation-approved thermal overpacks. The staff independently evaluated selected analyses performed by the applicant and concluded that they were acceptable (memorandum from R. Wescott to file, "Confirmatory Calculations for Fire Protection Review of National Enrichment Facility Integrated Safety Analysis Summary," dated March 22, 2005).

Hydrogen Control: Hydrogen is a highly flammable gas that can ignite or deflagrate at relatively low concentrations. It is used within the Technical Services Building (TSB) Chemical Laboratory and may be generated at battery-charging stations in the facility. The laboratory will be protected by one or more of the following features:

- Hydrogen piping will be provided with excess flow control;
- Hydrogen supply will be isolated by emergency shutoff valves interlocked with hydrogen detection in the areas served by the hydrogen piping; and
- Natural or mechanical ventilation will be provided to ensure that hydrogen concentrations do not exceed 25 percent of the lower explosive limit (LEL). If mechanical ventilation is provided, it will be continuous or will be interlocked to start on detection of hydrogen in the area. Mechanical ventilation will also be provided with airflow sensors, to sound an alarm if the fan becomes inoperative.

Hydrogen control in battery-charging stations will be provided by measures identified in National Fire Protection Association (NFPA) 70E, "Standard for Electrical Safety in the Workplace," and American National Standards Institute (ANSI) Standard C2, "National Electrical Safety Code."

Combustible Material Hazards: Materials of construction for the centrifuge process building, the supporting buildings, and centrifuge machines and components are predominantly noncombustible (e.g., steel, aluminum, concrete floors). A minimum of fixed combustibles is expected to be present in the operations areas, and the applicant plans to control transient combustibles to minimize potential fire hazards. The largest quantity of combustible material is two 19,000 L (5000 gal.) tanks of diesel fuel located outside.

Accident Scenarios

Staff reviewed the applicant's ISA Summary, which describes, qualitatively, the potential credible fire-accident scenarios and associated risks for the facility. The staff's evaluation is in Section 7.3.2 of the SER. The applicant postulated and evaluated the following key fire-accident scenarios:

- Fire in the Centrifuge Test Facility;
- Fire in the CRDB;
- Fire involving Cylinder Transporters/Movers
- Fire inside the Cascade Halls;
- Fire in the Process-Services Area;
- Fire inside the UF₆ handling area/Blending and Liquid Sampling Area;
- Fire inside the TSB; and
- Fire affecting the UBC Storage Pad.

The staff determined that the accident sequences identified by the applicant were sufficient to encompass all credible fire-related accident scenarios.

IROFS Related to Fire Safety

The NRC staff reviewed the applicant's identification of the required IROFS for preventing or mitigating fire accident scenarios that could lead to unacceptable performance in accordance with the requirements in 10 CFR 70.61. The staff's evaluation is in Section 7.3.3 of the SER. The applicant identified a set of IROFS that would ensure that the likelihood of a fire causing high consequence events is highly unlikely and the likelihood of a fire causing intermediate

consequence events is unlikely. Most of the IROFS for fire protection are administrative controls, such as for combustible loading control, or passive controls, such as fire barriers. The remaining fire protection features described in the Safety Analysis Report, such as the fire brigade, are measures that provide overall defense-in-depth protection. The staff determined that applicant's evaluation of accidents resulting from exterior and interior building explosions was acceptable and its conclusions that initiating events for these accident sequences would be highly unlikely, without the need for IROFS.

Facility Fire Protection

Staff reviewed the fire protection features to be used at the facility in Section 7.3.4 of the SER. Fire prevention at the facility consists of administrative controls to: (a) govern the handling of transient combustibles; (b) control ignition sources; (c) ensure that open flames or combustion-generated smoke is not used for leak-testing; (d) conduct periodic fire prevention inspections; (e) perform periodic house-keeping inspections; and (f) implement a system to control the disarming of fire-detection or fire-suppression systems.

In assessing the adequacy of fire protection for the facility, the staff considered construction of the buildings containing UF₆, all of which are of non-combustible pre-cast concrete-frame and concrete-panel construction.

The staff also evaluated the robustness of the fire alarm system, the placement of fire extinguishers, the sufficiency of the facility fire-water supply, and the design of wet pipe sprinkler systems to be installed.

The staff also reviewed the staffing, training, and design of the fire brigade, which will handle all minor fires and to provide a first-response effort designed to supplement the local fire department for major fires at the plant. The staff also reviewed the applicant's plan for training and coordinating with off-site assistance organization personnel.

8.0 EMERGENCY MANAGEMENT

The purpose of this review of the applicant's Emergency Management Plan is to determine if the applicant has established, before the start of operations, adequate emergency management facilities and procedures to protect workers, the public, and the environment. Emergency capability is incorporated into the baseline design criteria of 10 CFR Part 70, and is intended to ensure control of licensed material, evacuation of personnel, and availability of emergency facilities.

8.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

10 CFR 30.32(i)(1), 10 CFR 40.31(j)(1), and 10 CFR 70.22(i)(1)(i) specify when an applicant is not required to submit an Emergency Plan (EP) to the Nuclear Regulatory Commission (NRC). If an applicant is required to submit an EP, as described in 10 CFR 30.32(i)(1)(ii), 10 CFR 40.31(j)(1)(ii), and 10 CFR 70.22(i)(1)(ii), then the applicant must meet 10 CFR 30.32(i)(3), 10 CFR 40.31(j)(3), and 10 CFR 70.22(i)(3). In addition, 10 CFR 70.64(a)(6) requires applicants to address the control of licensed material, evacuation of personnel, and availability of emergency facilities for the design of new facilities. The acceptance criteria for NRC's review of the applicant's Emergency Management Plan are outlined in Section 8.4.3.1 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520.

8.2 STAFF REVIEW AND KEY ISSUES

8.2.1 Review Process

Staff reviewed the information provided in Chapter 8 of the Safety Analysis Report, the EP, and the Integrated Safety Analysis (ISA) Summary in accordance with guidance provided by NUREG-1520. The key information reviewed included: facility description; onsite and offsite emergency facilities; types, classification, and detection of accidents; mitigation of consequences; assessments of releases and responsibilities; notification and coordination and information to be communicated; training; safe shutdown; exercises and drills, and the responsibilities for maintaining the Emergency Program and its procedures current. In performing this review, the staff prepared a Request for Additional Information (RAIs) (letter to LES dated April 19, 2004), and resolved open items by reviewing the responses to the RAIs, dated May 19, 2004, and March 24, 2005. Staff also reviewed the applicant's baseline needs assessment of the response to fire and related emergencies, dated May 28, 2004. Based on the above communications with the staff, the applicant made appropriate revisions to the EP and ISA Summary.

8.2.2 Key Areas of Review

Facility Description

Staff reviewed the applicant's facility description in Section 8.3.1 of the Safety Evaluation Report (SER). Section 1.0 of the EP contains descriptions of the licensed activity, the facility and site, and the area near the site. This fulfills the requirements of 10 CFR 30.32(i)(3)(i), 10 CFR 40.31(j)(3)(i), and 10 CFR 70.22(i)(3)(i). The applicant will also maintain compliance with:

the *Emergency Planning and Community Right-to-Know Act of 1986*, in accordance with 10 CFR 30.32(i)(3)(xiii), 10 CFR 40(j)(3)(xiii), and 10 CFR 70(i)(3)(xiii).
Onsite and Offsite Emergency Facilities

Staff reviewed the applicant's onsite and offsite emergency facilities in Section 8.3.2 of the SER. Onsite emergency facilities are discussed in Sections 6.1 and 6.3 of the EP. The Control Room is the primary Emergency Operations Center (EOC). The designated alternate EOC is the Security Building, which is used depending on the nature and location of the emergency situation, or if the Control Room becomes uninhabitable. An offsite location will be available, if necessary. Offsite emergency support and equipment is discussed in Section 4.3 of the EP. Section 6.4 of the EP describes the emergency monitoring equipment that is available for personnel and area monitoring. Section 6.2 of the EP describes the communication systems that will be used at the facility.

Types, Classification, and Detection of Accidents

Staff reviewed the types, classification, and detection of accidents in Sections 8.3.3, 8.3.4, and 8.3.5 of the SER, respectively. In Section 2.1 of the EP, both postulated high- and intermediate-consequence events are identified. Accident sequences, as well as mitigating and preventive measures, are described. Nuclear criticality and loss of containment leading to a very large release of uranium hexafluoride (UF_6) are the only postulated events identified for which protective actions may be necessary. Staff concluded that this meets the requirements of 10 CFR 30.32(i)(3)(ii), 10 CFR 40.31(j)(3)(ii), and 10 CFR 70.22(i)(3)(ii).

Section 2.2 of the EP explains the methods and systems available to detect accidents at the facility. Actions in response to accidents are outlined in the Emergency Plan Implementing Procedures (EPIPs) and directed by the Shift Manager. The methods and systems presented in Section 2.2 of the EP fulfill the requirements of 10 CFR 30.32(i)(3)(iv), 10 CFR 40.31(j)(3)(iv), and 10 CFR 70.22(i)(3)(iv).

Section 3.1 of the EP explains the system used to classify an emergency as either an Alert or a Site-Area Emergency, and defines both types of incidents. The applicant stated that the threshold for escalating an event from an Alert to a Site-Area Emergency is based on indications of a release that could require a response by an offsite response organization, to protect persons offsite from reaching the offsite exposure limits set forth in 10 CFR 30.32(i)(1)(i), 10 CFR 40.31(j)(1)(i), and 10 CFR 70.22(i)(1)(i). This system for classifying events is acceptable to the staff and meets the requirements of 10 CFR 30.32(i)(3)(iii), 10 CFR 40.31(j)(3)(iii), and 10 CFR 70.22(i)(3)(iii).

Mitigation of Consequences

Staff reviewed the applicant's actions and equipment for mitigation of consequences in Section 8.3.6 of the SER. Section 5.3 of the EP describes actions and equipment that will be used to mitigate the consequences of accidents at the facility. The major hazard would be the chemical hazard caused by a release of UF_6 . The main features used at the facility to mitigate the consequences of accidents include automatic interruption or termination of specific operations, fire detection and suppression systems, operator response to abnormal conditions/alarms, and shutdown of the ventilation system, in case of a UF_6 release or a criticality. The actions,

features, and means for maintaining them, which are described in Sections 5.3 and 7.6 of the EP, fulfill the requirements of 10 CFR 30.32(i)(3)(v), 10 CFR 40.31(j)(3)(v), and 10 CFR 70.22(i)(3)(v).

Assessment of Releases and Responsibilities

Staff reviewed the applicant's assessment of releases and responsibilities in Sections 8.3.7 and 8.3.8 of the SER. Section 5.2 of the EP explains the actions that will be taken to assess the extent of an accident at the facility. In case of an Alert, dose projections of offsite radiation and hazardous material exposures will be made and provided to offsite response agencies. In addition, during a Site-Area Emergency, radiation or chemical surveys of the Assembly Area(s), the EOC, and the facility will be performed. Environmental sampling will be performed offsite, if necessary. These actions meet the requirements of 10 CFR 30.32(i)(3)(vi), 10 CFR 40.31(j)(3)(vi), and 10 CFR 70.22(i)(3)(vi).

The responsibilities of facility personnel during normal operations and during emergency situations are described in Sections 4.1 and 4.2, respectively, of the EP. The description of the responsibilities in the EP fulfills the requirements of 10 CFR 30.32(i)(3)(vii), 10 CFR 40.31(j)(3)(vii), and 10 CFR 70.22(i)(3)(vii).

Notification and Coordination and Information To Be Communicated

Staff reviewed the applicant's proposed notification and coordination and information to be communicated in Sections 8.3.9 and 8.3.10 of the SER. Section 3.2 of the EP provides a clear commitment to promptly notify offsite response organizations of an emergency, including notification to the NRC Operations Center. Sections 4.3 and 4.4 of the EP provide an adequate description of provisions for assistance from offsite response organizations. The staff concluded that this information meets the requirements of 10 CFR 30.32(i)(3)(viii), 10 CFR 40.31(j)(3)(viii), and 10 CFR 70.22(i)(3)(viii).

Section 3.3 of the EP provides an adequate description of the type of information to be given to offsite response organizations during an emergency. In the event of a Site Area Emergency, a standard recommendation will be provided to offsite assistance organizations with more specific data, as discussed in Section 3.3 of the EP (LES, 2005a). The staff determined that this meets the requirements of 10 CFR 30.32(i)(3)(ix), 10 CFR 40.31(j)(3)(ix), and 10 CFR 70.22(i)(3)(ix).

Training

In Section 8.3.11 of the SER, staff reviewed the applicant's emergency response training that will be provided to workers. Section 7.2 of the EP provides a description of the training the licensee will provide to workers on how to respond to an emergency. Emergency response personnel receive additional training annually, to provide specific information on how the emergency organization responds during emergency conditions, including staffing; determining and estimating potential offsite releases of radiation and chemicals; and interface with offsite assistance organizations. This training is required before workers are assigned to the emergency organization. Facility tours and classroom training are also provided to offsite response organizations. Each group will meet at least annually, with facility personnel, to accomplish this training and review items of mutual interest, including relevant changes to the

program. Refresher training is provided at least once every year. This fulfills the requirements of 10 CFR 30.32(i)(3)(x), 10 CFR 40.31(j)(3)(x), and 10 CFR 70.22(i)(3)(x).

Safe Shutdown

In Section 8.3.12 of the SER, staff reviewed the applicant's plans for safe shutdown of the facility. Section 9.1 of the EP states that during an emergency, immediate actions will be directed toward limiting the consequences of the incident to afford maximum protection to facility personnel and the general public. Once control of the facility has been reestablished, a systematic and planned approach to full facility recovery will be taken. Sections 5.2, 5.3, and 5.4 of the EP describes the methods to be used for assessing the extent of the event and the status of the facility, and the mitigative actions necessary to reduce or stop any ongoing releases of radioactive material or hazardous chemicals. Section 9.2 of the EP contains information regarding the staffing of the facility during the recovery phase of an event. This fulfills the requirements of 10 CFR 30.32(i)(3)(xi), 10 CFR 40.31(j)(3)(xi), and 10 CFR 70.22(i)(3)(xi).

Exercises and Drills

Staff reviewed the exercises and drills in Section 8.3.13 of the SER. Section 7.3 of the EP provides adequate provisions for drills, exercises, and biennial exercises that are used to test the adequacy of EIPs, emergency equipment, instrumentation, and to ensure all emergency response personnel are familiar and proficient with their duties. Offsite organizations are invited to participate in the biennial exercise, and NRC is invited to participate or observe. Exercise objectives and scenarios will be submitted to NRC, for review and comment, at least 60 days before the exercise. The information provided in Sections 7.3 and 7.4 of the EP meets the requirements of 10 CFR 30.32(i)(3)(xii), 10 CFR 40.31(j)(3)(xii), and 10 CFR 70.22(i)(3)(xii).

Responsibilities for Maintaining the Emergency Program and Its Procedures Current

In Section 8.3.14 of the SER, staff reviewed the applicant's responsibilities for developing and maintaining the emergency program and its procedures current. Section 7.1 of the EP explains that there will be site procedures for maintaining the EP and procedures current. Any proposed change to the EP that affects an offsite organization will be provided to that organization for review and comment at least 60 days before implementing the change, unless all parties mutually agreed otherwise. In accordance with 10 CFR 30.34(f), 10 CFR 40.35(f), and 10 CFR 70.32(i), the applicant may incorporate changes to the EP without receiving prior NRC approval, provided those changes do not decrease the effectiveness of the EP, and the NRC and affected offsite response organizations receive copies of the revised EP or procedures within 6 months of the changes.

9.0 ENVIRONMENTAL PROTECTION

The purpose of this review is to determine whether the applicant's proposed environmental protection measures are adequate to protect the environment, and the health and safety of the public.

9.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

To be considered acceptable, the applicant must satisfy the following regulatory requirements regarding environmental protection: 10 CFR Part 20, 10 CFR 30.33, 10 CFR 40.32(e), 10 CFR 51.60(b)(1)(vii), 10 CFR 70.22(a)(7), 10 CFR 70.59, and 10 CFR 70.65(b). The acceptance criteria for the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's environmental protection program are outlined in NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," Section 9.4.3.2.

9.2 STAFF REVIEW AND KEY ISSUES

9.2.1 Review Process

The staff reviewed the information provided in Chapter 9 of the applicant's Safety Analysis Report (SAR) and the Integrated Safety Analysis (ISA) Summary in accordance with guidance provided by NUREG-1520. The key information reviewed included air and liquid As Low As Reasonably Achievable (ALARA) goals, air and liquid effluent controls and monitoring, environmental monitoring, the ISA Summary. In performing this review, the staff prepared Requests for Additional Information (RAIs) (letter to Louisiana Energy Services, dated April 19, 2004), and resolved open items by reviewing the responses to the RAIs, dated May 19, 2004, and by a followup conference call with the applicant (memoranda from T. Johnson to J. Giitter, dated September 30, 2004). On March 14, 2005, the applicant provided additional clarifying information on its program for sampling airborne radioactive material. Based on above communications with the staff, the applicant made appropriate revisions to the SAR and ISA Summary.

9.2.2 Key Areas of Review

Air Effluent ALARA Goal

In Section 9.3.1.1 of the Safety Evaluation Report (SER), the staff reviewed the applicant's estimate of the maximum individual committed effective dose equivalent (CEDE) for air effluents during normal operations at the proposed facility and found that the CEDE would be less than 0.017 percent of the 1 mSv (100 mrem) limit on dose to the public in Part 20. The estimated maximum public dose was also found to be well below the 0.1 mSv (10 mrem) As Low As Reasonably Achievable (ALARA) constraint on air emissions described in 10 CFR 20.1101 (e.g., between 1 and 2 percent).

A reasonable initial ALARA goal for air effluents, described in NRC Regulatory Guides 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," and 8.37, "ALARA Levels for Effluents from Materials Facilities," is 10-20 percent of the regulatory limit.

The applicant will implement corrective actions that will likely result in doses that are a small fraction of the regulatory limits in 10 CFR Part 20 and 40 CFR Part 190. In addition, the calculated dose to the maximally exposed member of the public is a small fraction of the ALARA goals identified in the above-referenced Regulatory Guides. Therefore, staff found this initial estimate of air effluent quantities to be a reasonable ALARA goal for air effluent.

Liquid Effluent ALARA Goal

In Section 9.3.1.1 of the SER, the staff also reviewed the applicant's estimate of the maximum individual CEDE for liquid effluents during normal operations at the proposed facility. The CEDE to the maximally exposed member of the public would be less than 0.17 μ Sv (0.017 mrem) per year. The estimated maximum public dose is also well below the 0.1 mSv (10 mrem) ALARA constraint on liquid emissions described in 10 CFR 20.1101 (e.g., less than 2 percent).

A reasonable initial ALARA goal for liquid effluents, described in NRC Regulatory Guides 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," is 10-20 percent of the regulatory limit. The applicant committed to implement corrective actions that will likely result in doses that are a small fraction of the regulatory limits in 10 CFR Part 20 and 40 CFR Part 190. In addition, the calculated dose to the maximally exposed member of the public is a small fraction of the ALARA goals identified in Regulatory Guides 8.34 and 8.37. The applicant committed to implement corrective actions well below this level. Therefore, the staff found this initial estimate of liquid effluent to be a reasonable ALARA goal for liquid effluent.

Air Effluent Controls

In Section 9.3.1.2 of the SER, the staff evaluated the air effluent controls at the proposed facility, which include: (a) the Separations Building Gaseous Effluent Vent System (GEVS); (b) the Technical Services Building (TSB) GEVS; (c) the TSB Heating, Ventilating, and Air Conditioning (HVAC) system that services potentially contaminated areas; and (d) the Centrifuge Test and Post-Mortem Facilities Exhaust Filtration System. The staff found that the applicant had demonstrated that its air effluent controls will reduce releases to assure adequate protection of the environment and of health and safety of the public.

Liquid Effluent Controls

In Section 9.3.1.3 of the SER, the staff evaluated the liquid effluent controls at the proposed facility. These controls are for the Liquid Effluent Collection and Treatment System in the TSB. The applicant identified seven major sources of liquid waste from processes in the TSB and Separations Building. The dose to the maximally exposed offsite member of the public from liquid effluents is less than one percent of the regulatory limit. A reasonable initial ALARA goal for liquid effluent described in NRC Regulatory Guide 8.34 is 10-20 percent of the regulatory limit. Therefore, based on the above, the staff found that the applicant had demonstrated that it will reduce releases to adequately assure protection of the environment and of health and safety of the public.

Air Effluent Monitoring

In Section 9.3.2.1 of the SER, the staff evaluated the applicant's air effluent monitoring program and found that the expected concentrations of radioactive materials in airborne effluents would be well below the regulatory limits specified in 10 CFR 20.1302(c). The applicant proposed to demonstrate compliance with air effluent limits by calculation of the total effective dose equivalent to the individual who is likely to receive the highest dose. Such a demonstration of regulatory compliance is in accordance with 10 CFR 20.1302(b)(1). The staff reviewed the applicant's assumptions and conclusions used in its calculations and determined that they are reasonable.

On the basis of its analysis, including a review of specific applicant commitments, the staff found that the applicant's air effluent monitoring during operation of the facility will ensure that concentrations of radioactivity in air effluent are below the regulatory limits in 10 CFR Part 20.

Liquid Effluent Monitoring

In Section 9.3.2.2 of the SER, the staff evaluated the liquid effluent monitoring program proposed by the applicant. The expected concentrations of radioactive materials in liquid effluents are ALARA and are well below the limits specified in 10 CFR Part 20, Appendix B, Table 2. The applicant will demonstrate compliance with air effluent limits by calculation of the total effective dose equivalent to the individual who is likely to receive the highest dose in accordance with 10 CFR 20.1302(b)(1). The staff reviewed the applicant's assumptions and conclusions used in its calculations and determined that they are reasonable.

On the basis of its analysis, the staff found that the applicant's liquid effluent monitoring during operation of the facility ensures that concentrations of radioactivity in liquid effluent will be below the limits in 10 CFR Part 20.

Environmental Monitoring

In Section 9.3.2.4 of the SER, the staff evaluated the applicant's proposed environmental monitoring program. The applicant established its Radiological Environmental Monitoring Program (REMP), which is a major part of the applicant's effluent compliance program. The effectiveness of the applicant's effluent controls will be confirmed through implementation of the REMP. The purpose of the REMP is to verify confinement integrity at the facility and to support the primary means of demonstrating compliance with applicable radiation protection standards, which is effluent monitoring. The REMP sampling locations are based on NRC guidance found in NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors." The scope of the applicant's REMP meets the environmental monitoring criteria found in NUREG-1520, Section 9.4.3.3.2(2).

The applicant has an adequate and timely program to collect information to determine baseline concentrations of radionuclides, which information will be used to demonstrate compliance with applicable radiation protection standards. The staff finds that the applicant's environmental monitoring program adequately addresses applicable regulatory requirements and is acceptable.

ISA Summary

In Section 9.3.3 of the SER, the staff evaluated the applicant's compliance with the environmental performance requirements in 10 CFR 70.61(c)(3). The staff review of the ISA Summary confirms that the applicant has shown that it has adequately reduced the risks to the environment from accidents for which the consequences could otherwise exceed the environmental-consequence severity level.

The staff independently evaluated the accident sequences to identify whether a credible accident sequence could occur in which an environmental hazard could be created without also creating a worker-related hazard. Even when postulating conservative, multiple independent equipment failures combined with human error, the staff did not identify any accident sequence that would fail to meet the environmental performance requirements of section 10 CFR 70.61(c)(3). The applicant's approach to risk reduction is to be accomplished through a combination of preventive and mitigative measures, with an emphasis on preventive measures.

10.0 DECOMMISSIONING

The purpose of this review of the applicant's decommissioning plan is to determine that the applicant will be able to decommission the facility safely and in accordance with U.S. Nuclear Regulatory Commission (NRC) requirements. The purpose of NRC's review of the decommissioning funding plan (DFP) is to determine whether the applicant has considered decommissioning activities that may be needed in the future, has performed a credible site-specific cost estimate for those activities, and has presented NRC with financial assurance to cover the cost of those activities in the future.

10.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The following NRC regulations require planning, financial assurance, and record-keeping for decommissioning, as well as procedures and activities to minimize waste and contamination:

10 CFR 20.1401-1406	"Radiological Criteria for License Termination" (Subpart E)
10 CFR 30.35	"Financial Assurance and Recordkeeping for Decommissioning"
10 CFR 30.36	"Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas"
10 CFR 40.36(d)	"Decommissioning Funding Plan"
10 CFR 40.36	"Financial Assurance and Recordkeeping for Decommissioning"
10 CFR 40.42	"Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas"
10 CFR 70.22(a)(9)	"Decommissioning Funding Plan"
10 CFR 70.25	"Financial Assurance and Recordkeeping for Decommissioning"
10 CFR 70.38	"Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas"

The "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 and "Consolidated NMSS Decommissioning Guidance," NUREG-1757, define relevant regulatory guidance and appropriate acceptance criteria for decommissioning and DFPs contained in license applications.

10.3 STAFF REVIEW AND KEY ISSUES

10.3.1 Review Process

Staff reviewed the information provided in Chapter 10 of the applicant's Safety Analysis Report (SAR) in accordance with the guidance in NUREG-1757 and NUREG-1520. The key information reviewed was the applicant's conceptual decommissioning plan; the decommissioning cost estimate; and financial assurance for decommissioning. In performing this review, the staff prepared Requests for Additional Information (RAIs) (letters to the applicant dated April 19, 2004, and October 20, 2004, and an email to the applicant dated January 27, 2005) and resolved open items by reviewing responses to the RAIs (letters from the applicant dated May 12, 2004, May 19, 2004 (both classified and unclassified), June 18, 2004, December 10, 2004 (both classified and unclassified), January 7, 2005, March 3, 2005, March 29, 2005, April 8, 2005, May 11, 2005, May 16, 2005, and June 6, 2005), by follow-up

conference calls (memoranda from T. Johnson to J. Giitter, dated June 7, 2004, December 15, 2004, February 4, 2005, and April 11, 2005) and conducting in-office reviews (memoranda from T. Johnson to J. Clifford, dated April 29, 2005, and July 26, 2005). Based on the above communications, the applicant made appropriate revisions to the Safety Analysis Report.

10.2.2 Key Areas of Review

Conceptual Decommissioning Plan

In Sections 10.3.1.1 through 10.3.1.8, the staff evaluated the applicant's proposed conceptual decommissioning plan for the facility. In this review, the staff reviewed the applicant's plans for contamination control, worker exposure and waste volume control, the planned decommissioning steps, the decommissioning management and organization, health and safety, waste management, security of nuclear material, recordkeeping, and decontamination activities. The applicant's conceptual decommissioning plan is based on Urenco's experience using conventional decontamination techniques in decommissioning gas centrifuge enrichment plants in Europe.

The applicant's plan for decommissioning is to promptly decontaminate or remove all materials from the site, that prevent release of the facility for unrestricted use. This approach, referred to in the industry as DECON, avoids long-term storage and monitoring of wastes on site. The applicant estimates that the DECON alternative will take approximately 9 years to complete in three phases (3 years/module).

Decommissioning activities will generally include: (1) installation of decontamination facilities; (2) purging of process systems; (3) dismantling and removal of equipment; (4) decontamination and destruction of Confidential and Secret Restricted Data material; (5) sales of salvaged materials; (6) disposal of wastes; and (7) completion of a final radiation survey. Credit is not taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning.

The applicant states, in Section 10.1.6.7 of the SAR, that all wastes produced during decommissioning will be collected, handled, and disposed of in a manner similar to that described for those wastes produced during normal operation. Wastes will consist of normal industrial trash, non-hazardous chemicals and fluids, small amounts of hazardous materials, and radioactive wastes. The applicant states, in Section 10.1.6.7 of the SAR, that radioactive wastes will ultimately be disposed of in licensed low-level radioactive-waste disposal facilities. Hazardous wastes will be disposed of in permitted hazardous waste-disposal facilities. Non-hazardous and non-radioactive wastes will be disposed of in a manner consistent with good industrial practice and in accordance with all applicable regulations.

The applicant states, in Section 10.1.6.7 of the SAR, that Confidential and Secret-Restricted Data components and documents on site will be disposed of in accordance with the requirements of 10 CFR Part 95. Classified portions of the centrifuges will be destroyed; piping will likely be smelted; documents will be destroyed; and other items will be handled in an appropriate manner. Details will be provided in the facility "Standard Practice Procedures Plan for the Protection of Classified Matter and Information," submitted separately, in accordance with Part 95.

The applicant states, in Section 10.1.6.8 of the SAR, that it will perform a final radiation survey, to verify proper decontamination, to allow the site to be released for unrestricted use. The final survey report will include, among other things, a map of the survey site, measurement results, and the site's relationship to the surrounding area. If the results are above allowable residual radioactivity limits, further decontamination will be performed until the results are determined to be below limits.

Based on a review of the proposed conceptual decommissioning plan, the staff concluded that it was acceptable and was consistent with guidance in NUREG-1520 and NUREG-1757.

Decommissioning Costs

In Section 10.3.19 of the SER, the staff reviewed the applicant's decommissioning cost information. The applicant submitted decommissioning cost information consistent with the recommendations in NUREG-1757, Volume 3, "Consolidated NMSS Decommissioning Guidance - Financial Assurance, Recordkeeping, and Timeliness." The applicant presented its decommissioning cost estimate breakdown in SAR Tables 10.1-1 through 10.1-14. Decommissioning cost information included labor costs, proposed decontamination methods and unit costs, waste disposal costs, final survey costs, and costs for dispositioning depleted uranium tails. The decommissioning costs were based on the decommissioning experience of Urenco, the applicant's principal general partner, in decommissioning gas centrifuge enrichment plants in Europe.

The applicant estimates the cost of decommissioning the facility to be approximately \$942 million, in 2004 dollars, which includes an estimated cost of \$131 million to decommission the supporting structures, an estimated tails-disposition cost of \$622 million, and a 25 percent contingency factor, equal to \$188 million.

The applicant conservatively estimated that the facility will generate 132,942 MT of depleted uranium over a nominal 30 years of production, and did not reduce the estimate of depleted uranium based on the planned operations approach where production would actually end 5 years earlier. The applicant estimated the waste processing and disposal cost of UF_6 tails at \$4.68 per kilogram of uranium (kg U) or \$4,680 per metric ton of uranium (MTU). This cost is based on the total of the three cost components that make up the total disposition cost for DUF_6 (i.e., deconversion, disposal, and transportation). The staff reviewed the basis of each of these three cost components, and has concluded that they are reasonable.

The deconversion cost was based on proprietary information on a previously proposed private deconversion plant using the Cogema dry conversion process producing U_3O_8 and aqueous hydrogen fluoride (HF). The proposed process was the same as the plant Cogema has been operating in Pierrelatte, France for 20 years.

The Cogema proposal assumed that HF would be sold commercially and did not include the costs to neutralize aqueous HF to calcium fluoride. Staff consider that neutralization would have no effect on the overall deconversion costs because those costs would be balanced by the elimination of costs for equipment for storing HF prior to commercial sale. The cost of disposing the calcium fluoride (\$0.02/kg U) was included in the estimate.

The transportation and disposal costs were based on estimates provided by vendors of transportation and disposal services. Transportation costs were based on an estimate from Transportation Logistics International. This transportation estimate (\$0.85/kg U) was independent of distance. The disposal cost of \$1.14/kg U for depleted uranium oxides was based on an estimate provided by Waste Control Specialists. Staff compared the Waste Control Specialists estimate to an estimate for disposal of decommissioning wastes the applicant had obtained from Envirocare of Utah and found it to be consistent. The Envirocare disposal estimate for decommissioning waste was \$2.12/m³ (\$75/ft³) (LES, 2004). For the disposal of U₃O₈, the equivalent disposal cost at Envirocare is \$1.07/kg U.

Further, the applicant submitted an estimate for tails disposition from the U.S. Department of Energy (DOE) (DOE, 2005) as additional evidence of the reasonableness of their estimate. The DOE estimate included conversion, transportation, storage, disposal, and decommissioning costs of the conversion facility and totaled \$3.34/kg DUF₆ (\$4.91/kg U) in 2004 dollars. This is less than 5 percent of the difference in the applicant's estimate of \$4.68/kg U. Staff considers that the DOE estimate provides additional assurance that the applicant's estimate of depleted uranium disposition costs is reasonable.

Based on the staff's review of the classified and unclassified information, the staff found that the cost estimate for decommissioning the facility is reasonable and the cost estimate fulfills the requirements of 10 CFR 30.35(e), 10 CFR 40.36(d), and 10 CFR 70.25(e) and the evaluation criteria in Section 4.1 of NUREG-1757, Volume 3.

Financial Assurance for Decommissioning

In Section 10.3.1.10 of the SER, the staff reviewed the applicant's proposed financial assurance mechanism. The applicant stated it will utilize a surety bond method to provide reasonable assurance of decommissioning funding as required by 10 CFR 30.35(f)(2), 10 CFR 40.36(e)(2), and 10 CFR 70.25(f)(2). The applicant provided draft copies of the surety bond and standby trust language. Finalization of the specific financial instruments to be utilized will be completed, and signed originals of those instruments will be provided to the NRC for final confirmation of the instrument prior to the applicant receiving licensed material at the facility.

The surety bond method to be adopted by the applicant will provide a guarantee that decommissioning costs will be paid in the event the applicant is unable to meet its decommissioning obligations at the time of decommissioning. The surety bond will be structured consistent with applicable NRC requirements and in accordance with NRC regulatory guidance contained in NUREG-1757, Volume 3.

In accordance with 10 CFR 30.35(e), 10 CFR 40.36(d), and 10 CFR 70.25(e), the applicant will update the decommissioning cost estimate for the facility and the associated funding levels, over the life of the facility. These updates will take into account changes resulting from inflation or site-specific factors, such as changes in facility conditions or expected decommissioning procedures. These funding level updates will also address anticipated accumulated tails. As required by 10 CFR 30.35(e), 10 CFR 40.36(d), and 10 CFR 70.25(e), such updates will occur at least every 3 years.

In Section 1.2.5 of the SAR, the applicant requested an exemption to fund decommissioning on an incremental basis. Section 1.2.3.6 of the SER discusses the approval of this exemption request as required in 10 CFR 40.14 and 10 CFR 70.17. The applicant's proposed schedule for updating the decommissioning cost estimate and financial assurance instruments will provide updates at a frequency of at least every 3 years in accordance with 10 CFR 30.35(e), 10 CFR 40.36(d), and 10 CFR 70.25(e). This approach to funding the financial assurance instrument is acceptable to NRC staff because the amount of financial assurance will be sufficient to cover the decommissioning obligation of the licensee at any point in time in the event that the licensee is unable to complete decommissioning for any reason.

Because final executed copies of the financial assurance mechanism will not be provided to NRC until prior to receipt of licensed material, NRC staff is imposing license conditions to ensure that the applicant will provide executed financial assurance instruments prior to receipt of licensed material and that updates to the financial assurance amounts are provided consistent with the applicant's commitments. With these proposed license conditions and the exemption discussed in Section 1.2.3.6 of the SER, NRC staff concluded that the DFP and proposed surety bond method acceptable.

11.0 MANAGEMENT MEASURES

The purpose of this review is to verify whether the applicant provided information to ensure that the management measures applied to items relied on for safety (IROFS), as documented in the Integrated Safety Analysis (ISA) Summary, provide assurance that the IROFS will be available and reliable, consistent with the performance requirements of 10 CFR 70.61. If a graded approach is used, the review should also determine whether the measures are applied to the IROFS in a manner commensurate with the IROFS' importance to safety.

11.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The regulatory basis for the review of management measures is governed by 10 CFR 19.12, 70.4, 70.22(a)(8), 70.62(a)(3), 70.62(d), 70.72, and 70.74(a) and (b). The acceptance criteria the U.S. Nuclear Regulatory Commission (NRC) uses for review of management measures are outlined in Section 11.4.3 of NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility."

11.2 STAFF REVIEW AND KEY ISSUES

11.2.1 Review Process

NRC staff reviewed the applicant's Quality Assurance Program Description (QAPD) and information provided in Chapter 11 of the applicant's Safety Analysis Report (SAR) and ISA Summary in accordance with the guidance provided in NUREG-1520. The key information reviewed included the application of the following management measures applied to IROFS: configuration management (CM); maintenance; training and qualifications; procedure development and implementation; audits and assessments; incident investigations; records management; and other quality assurance (QA) elements.

Staff reviewed the applicant's Quality Assurance Program Description (QAPD) that was submitted by the applicant on December 3, 2002, prepared requests for additional information (RAIs) (letter to the applicant dated March 25, 2003), and resolved open items by reviewing responses to the RAIs dated November 21, 2003, and March 16, 2004, and by a followup conference call with the applicant (memorandum from Y. Faraz to J. Giitter conference call summary dated April 3, 2003). The QAPD was approved by the U.S. Nuclear Regulatory Commission (NRC) on April 9, 2004. The staff also reviewed Chapter 11 of the SAR, prepared RAIs (letter to the applicant dated April 19, 2004), and resolved open items by reviewing responses to the RAIs dated May 19, 2004, and a followup conference call with the applicant (memorandum from T. Johnson to J. Giitter, dated April 11, 2005). Based on the above communications with the staff, the applicant made appropriate revisions to the Safety Analysis Report and ISA Summary.

11.2.2 Key Areas of Review

Configuration Management

In Section 11.3.1 of the Safety Evaluation Report (SER), staff evaluated the applicant's commitment to develop and implement a CM to ensure accurate, current documentation that

matches the facility's physical/functional configuration, and ensures that IROFS are available and reliable, and comply with regulatory requirements. The CM program will be implemented throughout facility design, construction, testing, and operation. In all cases, the applicant's CM program will provide for the control of key documentation, including the ISA, in accordance with design control, document control, and records management procedures. All design changes will undergo formal review, including interdisciplinary reviews, as appropriate. The CM program will be applied to all structures, systems, and components (SSCs) that the ISA identifies as IROFS, and any items that affect the function of the IROFS. The applicant provided a descriptive review process which will be implemented to ensure changes to the facility's physical/functional configuration, procedures, and controlled documents are implemented in accordance with the provisions of 10 CFR 70.72. Changes to the above items will be controlled through procedures. The staff's evaluation found that the applicant has suitably and acceptably described its implementation strategy for a CM program that meets the requirements of 10 CFR 70.72, and the CM program appropriately covered CM policy, design requirements, document control, change control, and assessments. The staff concluded that the applicant's CM program is acceptable.

Maintenance

In Section 11.3.2 of the SER, the staff reviewed the applicant's proposed maintenance and functional testing programs. The applicant outlined the planned and scheduled maintenance and functional testing programs, for IROFS, that will ensure that equipment and controls will be maintained in a condition of readiness and will perform their safety functions when required. The maintenance organization plans, schedules, and tracks maintenance activities, and maintains records for these activities. The applicant described implementing measures that ensure that the quality of facility SSCs will not be compromised by planned modifications or maintenance activities. The applicant described a set of methods and practices that will be applied to its corrective maintenance, preventive maintenance, and functional-test maintenance elements. The staff determined that the applicant's maintenance functions meet the requirements of 10 CFR Part 70 and provide assurance of protecting the health and safety of workers, the public, and the environment.

Training and Qualifications

In Section 11.3.3 of the SER, staff evaluated training, testing, and qualification of personnel who perform activities relied on for safety. Operations phase training programs include training for pre-operational functional testing and initial-start-up testing. The applicant states that qualification will be indicated by successful completion of prescribed training, demonstration of the ability to perform assigned tasks, and, where required by regulation, maintaining an active and valid license issued by a regulatory agency. The applicant described the methods that will be employed to perform an analysis of job training needs and to ensure that personnel who work on tasks related to IROFS are provided the appropriate training. The staff's evaluation found that the applicant's description of its training program appropriately covered: (a) training organization and management; (b) analysis and identification of functional and position training requirements; (c) training basis and objectives; (d) organization of instruction; (e) evaluation of trainee learning; (f) conduct of on-the-job training; (g) evaluation of training program effectiveness; (h) personnel qualification; and (l) personnel evaluations. Based on this evaluation, the staff finds the applicant's training program acceptable.

Procedure Development and Implementation

In Section 11.3.4 of the SER, staff reviewed the applicant's processes for developing and implementing procedures planned and written for the operation of IROFS and for all management measures supporting those IROFS. All applicant activities involving licensed materials or IROFS will be conducted in accordance with approved procedures. The applicant identified four types of plant procedures that will be used to control activities throughout the facility: (1) operating procedures, (2) administrative procedures, (3) maintenance procedures, and (4) emergency procedures. The identification of needed procedures will take into consideration the results of the ISA. The staff determined that the applicant described a suitably detailed process for the development, approval, and implementation of procedures. The staff concludes that the applicant's strategic plan for procedure development will meet the requirements of 10 CFR Part 70.

Audits and Assessments

In Section 11.5 of the SAR, staff reviewed the applicant's policies, directives, and commitments to conduct internal audits and independent assessments of activities significant to facility safety and environmental protection. The applicant described a system of audits and assessments that consists of two distinct levels of activities: an audit activity structured to monitor compliance with regulatory requirements, licensing commitments, and facility procedures; and an assessment activity oriented toward determining the effectiveness of the activities in ensuring that IROFS are reliable and available to perform their intended safety functions. Audits and assessments will be conducted for the following areas: (a) nuclear criticality safety, radiation safety, and chemical safety; (b) industrial safety and fire protection; (c) environmental protection; (d) emergency management; (e) QA; (f) CM; (g) maintenance; (h) training and qualification; (i) procedures; (j) corrective action program (CAP) and incident investigation; and (k) records management. The staff concluded that the applicant's plan for audits and assessments meets the requirements of 10 CFR Part 70 and provides assurance of protection of the health and safety of workers, the public, and the environment.

Incident Investigations

In Section 11.3.6 of the SER, the staff reviewed the applicant's policy, procedures, and management structure used to investigate abnormal events and completing appropriate corrective actions. The applicant's overall incident investigation process will provide for incident identification, investigation, root-cause analysis, environmental protection analysis, recording, reporting, follow-up, and reporting events to NRC, as required by 10 CFR 70.50 and 70.74. These activities will be performed according to written corrective action process procedures. Each event or condition will be evaluated to determine the level of investigation required. Guidance for evaluating the significance of occurrences will be contained in corrective action process procedures, and the extent of the investigation will depend on the significance of the incident, with respect to the levels of uranium released or the potential for exposure to workers, the public, or the environment. The staff concluded that the applicant's incident investigation process complies with applicable NRC regulations, is consistent with NUREG-1520, and provides assurance of protection of the health and safety of workers, the public, and the environment.

Records Management

In Section 11.3.7 of the SER, staff reviewed the applicant's records management system for handling and storing health and safety records and the records generated or needed in the design, construction, operation, and decommissioning phases of the facility. QA records will include all completed records that furnish documentary evidence of the quality of items or activities affecting quality, including records related to health and safety. The applicant will use a records storage system that is capable of protecting and preserving health and safety records that are stored at the facility during the mandated periods, as well as, the capabilities of the storage system for protecting stored records from loss, theft, tampering, or damage, during and after emergencies. The staff concluded that the applicant's records management system is acceptable.

Other QA Elements

In Section 11.3.8 of the SER, staff reviewed other QA elements to obtain reasonable assurance of the implementation of accepted QA principles applied to management measures during the design, construction, operation, maintenance, and modification phases of the facility's life. Staff reviewed the applicant's complete description of the application of QA elements applied to IROFS.

The applicant included a complete description of its application of QA elements to IROFS. The applicant will assign QA levels to facility SSCs and associated processes, based on their safety significance. Each component and document will receive a categorization of QA Level 1, 2, or 3. The applicant's QA Program and its supporting manuals, procedures, and instructions are applicable to items and activities designated as QA Levels 1 and 2. QA Level-1 will conform to the criteria established in 10 CFR Part 50, Appendix B. These criteria will be met by conformance with American Society of Mechanical Engineers (ASME) NQA-1-1994, including supplements as revised by the ASME NQA-1a-1995 Addenda. The QA Level-1 Program will be applied to those SSCs and administrative controls that have been determined to be IROFS; items that affect the functions of IROFS; and items required to satisfy those regulatory requirements applicable to QA Level-1. The applicant's QA Level-2 Program is an owner-defined QA Program that uses the ASME NQA-1-1994 standard, including supplements as revised by the ASME NQA-1a-1995 Addenda, as guidance.

The staff concluded that the applicant's application of other QA elements meets the requirements of 10 CFR 70.62(d), and other applicable regulations (Appendix B of 10 CFR Part 50), and provides the assurance of protection of public health and safety and protection of the environment.

PROPRIETARY INFORMATION

12.0 MATERIAL CONTROL AND ACCOUNTING

The purpose of this review was to verify that the applicant, Louisiana Energy Services (LES), provided sufficient information, in its Fundamental Nuclear Material Control Plan (FNMCP) (LES, 2004) to determine that the Material Control and Accounting (MC&A) program meets the applicable regulatory requirements in 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material."

12.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The FNMCP will be considered acceptable if it meets the regulatory requirements and guidance specified in: 1) 10 CFR Part 74; 2) NUREG/CR-5734, "Recommendations to the NRC on Acceptable Standard Format and Content for the Fundamental Nuclear Material Control (FNMCP) Plan Required for Low-Enriched Uranium Enrichment Facilities;" 3) Regulatory Guide 5.67, "Material Control and Accounting Requirements for Uranium Enrichment Facilities Authorized to Produce Special Nuclear Material of Low Strategic Significance;" NUREG/BR-0006, "Instructions for Completing Nuclear Material Transaction Reports (DOE/NRC Forms 74-1 and 740M);" 5) NUREG/BR-0007, "Instructions for the Preparation and Distribution of Material Status Reports (DOE/NRC Forms 742 and 742C);" and 6) NUREG/BR-0096, "Instructions and Guidance for Completing Physical Inventory Summary Reports (NRC Form 327)."

12.2 STAFF REVIEW AND KEY ISSUES

12.2.1 Review Process

The staff reviewed and evaluated information provided by the applicant in the FNMCP for the proposed MC&A program in accordance with guidance provided in NUREG/CR-5734. The key information reviewed was the MC&A organization, the measurement program, the statistical program, the physical inventory program, the item control program, the material receipt and shipment program, the assessment program, the unauthorized enrichment prevention program, the program for resolving indications of missing uranium, investigation assistance, and record-keeping. In performing this review, the staff prepared Requests for Additional Information (RAIs) (letter to LES dated April 19, 2004), by reviewing responses to RAIs (letters from LES dated May 19, 2004, and July 30, 2004), and by followup conference calls with the applicant (memoranda from T. Johnson to J. Giitter, dated August 2, 2004, and September 23, 2004). Based on the above communications, the applicant made appropriate revisions to the FNMCP.

12.2.2 Key Areas of Review

MC&A Organization

In Section 12.3.1 of the SER, the staff reviewed the applicant's management structure and organization positions that have responsibilities related to the MC&A program. The staff found that the applicant has an organization, position qualification requirements, core procedures, and a training program capable of effective management of the MC&A system and that the MC&A

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program performance will not be adversely affected by the facility management structure, as required in 10 CFR 74.33(c)(1).

Measurement Program

In Section 12.3.2 and 12.3.3 of the SER, the staff reviewed the measurements that will be used for accounting purposes or for a monitoring program to detect an unauthorized activity and the measurement control program to ensure that measurements are properly made. The staff found the applicant's system of measurements appropriate and acceptable to ensure that all quantities of nuclear material in the accounting records are based on reliable measurements, as required in 10 CFR 74.33(c)(2). The staff also concluded that the FNMCP adequately described a measurement control program that ensures the capabilities required in 10 CFR 74.33(c)(3) are met.

Statistical Program

In Section 12.3.4 of the SER, the staff reviewed the applicant's statistical program to evaluate MC&A data. The FNMCP detailed the procedures and methods for statistically evaluating MC&A data. The staff found that the statistical program described is adequate.

Physical Inventory Program

In Section 12.3.5 of the SER, the staff reviewed the applicant's basic elements for scheduling, performing, and reporting bimonthly and annual physical inventories. The staff found that the FNMCP demonstrates its ability to confirm the presence and quantities of nuclear materials, as required in 10 CFR 74.33(c)(4).

Item Control Program

In Section 12.3.6 of the SER, the staff reviewed the applicant's overall MC&A system's ability to maintain a record of all source material (SM) and special nuclear material (SNM). The staff found that the FNMCP adequately described an item control program that will identify all SNM and SM contained in all items, as required in 10 CFR 74.33(c)(6).

Material Receipt and Shipment Program

In Section 12.3.7 of the SER, the staff reviewed the applicant's proposed practices and methods for receipt, transfer, and shipment of nuclear materials and evaluation and resolution of shipper-receiver differences and found them appropriate and acceptable with regard to the resolution program contained in 10 CFR 74.33(c)(7).

Assessment Program

In Section 12.3.8 of the SER, the staff reviewed the applicant's independent assessment program for review of the facility MC&A program capabilities, performance, and overall

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effectiveness on a periodic basis. The staff found that the program will provide for documented independent assessments within the required 24 months and conforms to the requirements contained in 10 CFR 74.33(c)(8).

Unauthorized-Enrichment Prevention Program

In Section 12.3.9 of the SER, the staff reviewed the applicant's program for precluding and detecting unauthorized production of enriched uranium. The staff determined that the approaches and methods for the detection, resolution, and reporting programs comply with 10 CFR 74.33(c)(5).

Program for Resolving Indications of Missing Uranium

In Section 12.3.10 of the SER, the staff reviewed the applicant's proposed methods and procedures for resolving indicators of missing nuclear materials or unauthorized production of enriched uranium. The staff determined that the approaches and methods of the detection, resolution, and reporting programs comply with the requirements of 10 CFR 74.33(a).

Investigation Assistance

In Section 12.3.11 of the SER, the staff reviewed the applicant's program to provide informational items to the U.S. Nuclear Regulatory Commission or other Government agencies to assist in any investigation relating to actual or highly suspicious events pertaining to missing uranium or unauthorized enrichment. The staff found that the commitments in the FNMCP meet the requirements in 10 CFR 74.33(a) to provide information to aid in the investigation of missing uranium and unauthorized production activities.

Record-keeping

In Section 12.3.12 of the SER, the staff reviewed the applicant's nuclear material accounting and record-keeping system. The staff found that the accounting system is secure and adequate and identified the necessary documentation needed. The staff determined that the applicant's record-keeping system is acceptable and conforms to the requirements in 10 CFR 74.33(d).

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13.0 PHYSICAL PROTECTION

The purpose of this review was to verify that the applicant, Louisiana Energy Services (LES), provided sufficient information to conclude, with reasonable assurance, that there is an adequate physical protection plan for special nuclear material (SNM) of low strategic significance at the proposed uranium enrichment facility to be located in Lea County, New Mexico.

13.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The applicant's Physical Security Plan for the protection of SNM of low strategic significance will be considered acceptable if it meets the regulatory requirements and guidance specified in 10 CFR 73.67, 10 CFR 73.71, and Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate to Low Strategic Significance."

13.2 STAFF REVIEW AND KEY ISSUES

13.2.1 Review Process

Staff reviewed the information provided in the applicant's Physical Security Plan in accordance with guidance provided by Regulatory Guide 5.59. The key information reviewed was in the areas of barriers, access control, intrusion detection, response force, and event reporting. In performing this review, the staff prepared Requests for Additional Information (RAIs) (letter to LES dated April 19, 2004), reviewed responses to RAIs (letters from LES dated May 12, 2004, and July 30, 2004), and by followup conference calls with the applicant (memoranda from T. Johnson to J. Giitter, dated July 13, 2004, and September 10, 2004). Based on the above communications with the staff, the applicant made appropriate revisions to the Physical Security Plan.

13.2.2 Key Areas of Review

Barriers

In Section 13.3.1 of the SER, the staff reviewed the applicant's proposed barriers for physical protection of the facility. [REDACTED]
[REDACTED]
[REDACTED] The staff concluded that this barrier system meets the requirement in 10 CFR 73.67.

Access Control

In Sections 13.3.3 and 13.3.4 of the SER, the staff reviewed the applicant's proposed personnel and vehicle access control measures. [REDACTED]
[REDACTED]

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[REDACTED]

[REDACTED]

The staff reviewed the access control measures for personnel and vehicles are found that the measures comply with 10 CFR 73.67.

Intrusion Detection

In Section 13.3.5 of the SER, the staff reviewed the applicant's proposed intrusion detection measures. [REDACTED]

[REDACTED] The staff determined that these measures meet the requirements in 10 CFR 73.67.

Response Force

In Section 13.3.6 of the SER, the staff reviewed the applicant's proposed response to unauthorized penetrations into the facility. [REDACTED]

[REDACTED] The staff concluded that the response force meets the requirements in 10 CFR 73.67.

Event Reporting

In Section 13.3.7 of the SER, the staff reviewed the applicant's plan for reporting security events. [REDACTED]

The staff concluded that the proposed reporting is in accordance with 10 CFR 73.71.

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**14.0 PHYSICAL SECURITY OF THE
TRANSPORTATION OF SPECIAL NUCLEAR MATERIAL
OF LOW STRATEGIC SIGNIFICANCE**

The purpose of this review is to verify that the applicant, Louisiana Energy Services (LES), provided sufficient information to conclude, with reasonable assurance, that there is an adequate physical protection plan for the transportation of special nuclear material of low strategic significance (SNM-LSS) to, or from, the applicant's proposed uranium enrichment facility to be located in Lea County, New Mexico.

14.1 REGULATORY REQUIREMENTS AND REGULATORY ACCEPTANCE CRITERIA

The regulatory requirements for physical protection of SNM-LSS in transit are specified in 10 CFR 73.67(c); 73.67(g)(1) - (5); 73.71; 73.73; 73.74; and 74.15. The applicant's physical security plan for the transportation of SNM-LSS will be considered acceptable if it meets the regulatory requirements specified in 10 CFR 73.67; 73.71; 73.73; 73.74; and 74.15.

14.2 STAFF REVIEW AND KEY ISSUES

14.2.1 Review Process

Staff reviewed the information provided in Chapters 7-10 (transportation provisions) of the applicant's Physical Security Plan. This key information reviewed identified the licensee's commitments as they address the regulatory requirements related to material transportation, material receipt, in-transit physical protection, exports, imports, and document retention. The information was reviewed against the regulatory requirements specified in 10 CFR 73.67 (c), 73.67(g)(1) - (5), 73.71, 73.73, and 73.74. The review also was informed by Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate and Low Strategic Significance," and Regulatory Guide 5.15, "Tamper-Indicating Seals for the Protection and Control of Special Nuclear Material."

In performing the review, the staff prepared Requests for Additional Information (RAIs) (letter to LES dated November 24, 2004). The staff discussed transportation security issues with LES officials in a meeting, which took place at the NRC on November 10, 2004 (memorandum from T. Johnson to J. Giitter, dated November 18, 2004). The staff resolved open items by conducting conference calls with the applicant (memoranda from T. Johnson to J. Giitter, dated December 2, 2004 and January 13, 2005), and by reviewing the responses to the RAIs, dated December 10, 2004, and January 12, 2005. Based on the above communications with the staff, the applicant made appropriate revisions to the Physical Security Plan.

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7.2.2 Key Areas of Review

The staff evaluated the portion of the applicant's Physical Security Plan relating to the transportation of SNM-LSS shipments to and from the facility in Chapter 14.0 of the Safety Evaluation Report (SER).

Material Transportation Requirements

In Section 14.3.1 of the SER, the staff evaluated the Physical Security Plan to confirm the licensee's commitments as a shipper. Prior to shipment of SNM-LSS, LES will notify the receiver of the mode of transportation, estimated time of arrival, location of SNM-LSS transfer point, name of carrier, and transport identification. LES will not release a shipment from the site without receiver's acknowledgment of its readiness to receive the shipment. Every container will be properly sealed with a security seal (tamper-indicating device) and inspected just prior to the commencement of the shipment. Finally, LES will assure that the responsibility for the in-transit physical protection of SNM-LSS will be designated in advance, in writing, to either LES or the licensee receiver.

Receiver Requirements – Transportation

In Section 14.3.2 of the SER, the staff evaluated the Physical Security Plan to confirm the licensee's commitments as a receiver. Upon receipt of a shipment at the facility, LES will verify the integrity of all security seals. LES will provide a notification of received shipment within ten days to the shipper. LES will either acknowledge responsibility for the in-transit physical protection of SNM-LSS or will ensure that a prior written agreement from the shipper has been received in which the shipper accepts either full or shared responsibility for the in-transit physical protection of SNM-LSS in accordance with 10 CFR 73.67(g)(2)(iii) and 10 CFR 73.67(g)(3).

In-Transit Physical Protection Requirements

In Section 14.3.3 of the SER, the staff reviewed the response procedures for dealing with threats of theft or thefts of SNM-LSS while in transit. The staff evaluated the Physical Security Plan to confirm that arrangements will be made for notifying the licensee, who arranges for the in-transit protection of SNM-LSS, of the arrival of the shipment to its destination. The staff confirmed that if any shipment of SNM-LSS is determined to be lost or unaccounted for, the shipper will launch a trace investigation, and the Nuclear Regulatory Commission (NRC) will be notified in accordance with 10 CFR 73.71.

Export Requirements

In Section 14.3.4 of the SER, the staff reviewed the licensee's commitment to protecting export shipments of SNM-LSS. SNM-LSS in transit outside the United States will be protected in accordance with Annex I of the Convention of Physical Protection of Nuclear Material as required under 73.73(a)(4). For the domestic U.S. portion of the shipment, the licensee will

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comply with the transportation security requirements identified in 10 CFR 73.67(g)(1) and (3). The applicant's plan also contains procedures for making advanced notifications to the NRC of export shipments of SNM-LSS that originates from the facility, as required under 10 CFR 73.73(a)(1-3).

Import Requirements

In Section 14.3.5 of the SER, the staff reviewed the licensee's commitment to protecting import shipments of SNM-LSS. LES will comply with the requirements of 10 CFR 73.67(g)(2) and (3) for shipments from a country that is a party to the Convention on the Physical Protection of Nuclear Material. For a country that is not a party to the Convention on the Physical Protection of Nuclear Material, LES will protect any such shipment in accordance with Annex I to the Convention on the Physical Protection of Nuclear Material during its transport outside the United States, as required under 10 CFR 73.74(c), and in accordance with 10 CFR 73.67(g)(2) and (3) during its transit within the United States (i.e., from the port of entry). LES will notify the person or customer who delivered the material to a carrier for transport of the arrival of such material at the facility as required under 10 CFR 73.67(g)(5)(ii). The staff applicant's plan also contains procedures for making advance notifications to NRC of import shipments of SNM-LSS from countries that are not party to the Convention on Physical Protection of Nuclear Material, as required under 10 CFR 73.74(a)(1-3).

Document Retention Requirements

The staff confirmed the licensee's commitment to meeting the security document retention requirements specified in 10 CFR 73.67(c)(2), 73.67(g)(3)(I), 73.67(g)(4), and 73.67(g)(5)(I).

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