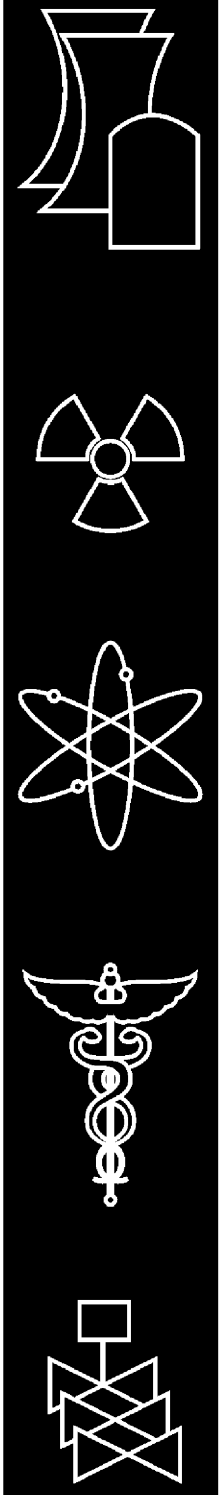
NUREG-1851



Safety Evaluation Report
for the American Centrifuge Plant
in Piketon, Ohio

Docket No. 70-7004

USEC, Inc.

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

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ABSTRACT

The report documents the U.S. Nuclear Regulatory Commission (NRC) staff review and safety and safeguards evaluation of USEC Inc.'s (USEC's) application for a license to construct a gas centrifuge uranium enrichment facility and possess and use special nuclear material (SNM), source material, and byproduct material. USEC proposes that the gas centrifuge uranium enrichment facility be located on the U.S. Department of Energy's (DOE's) reservation in Piketon, Ohio. The facility will possess natural, depleted, and enriched uranium, and will be authorized to enrich uranium up to a maximum of 10 weight percent uranium-235.

The objective of this review is to evaluate the facility's potential adverse impacts on worker and public health and safety, under both normal operating and accident conditions. The review also considers physical protection of SNM and classified matter, material control and accounting of SNM, and the management organization, administrative programs, and financial qualifications provided to ensure safe design and operation of the facility.

NRC staff concludes, in this safety evaluation report, that the applicant's descriptions, specifications, and analyses provide an adequate basis for safety and safeguards of facility operations and that operation of the facility does not pose an undue risk to worker and public health and safety.

Potential environmental impacts associated with the proposed facility and its reasonable alternatives are addressed in a separate document, Environmental Impact Statement for the Proposed American Centrifuge Plant in Piketon, Ohio, NUREG-1834, which was made available to the public in April 2006.

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EXECUTIVE SUMMARY

On August 23, 2004, USEC Inc. (the applicant) submitted, to the U.S. Nuclear Regulatory Commission (NRC), an application requesting a license, under 10 CFR Parts 30, 40, and 70, to possess and use byproduct, source, and special nuclear material (SNM) in a gas centrifuge uranium enrichment facility. The applicant proposes that the facility be located on the U.S. Department of Energy (DOE) reservation in Piketon, Ohio, and have a nominal capacity of 3.5 million separative work units (SWUs). (A SWU is a unit of enrichment that measures the effort required to separate isotopes of uranium). The facility will possess natural, depleted, and enriched uranium, and will enrich uranium up to a maximum of 10 percent uranium-235. The applicant also requested a facility clearance for classified information, under 10 CFR Part 95.

NRC staff conducted its safety review in accordance with NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." The staff's safeguards review involved reviews of the applicant's Fundamental Nuclear Material Control Plan (FNMCP); the Physical Security Plan, which includes transportation security; and a "Standard Practice Procedures Plan for the Protection of Classified Matter." The staff also reviewed the applicant's Quality Assurance Program Description and Emergency Plan. Where the applicant's design or procedures should be supplemented, NRC staff has identified license conditions to provide assurance of safe operation.

In conducting its safety review, the staff assessed, among other things, whether the applicant's proposed equipment, facilities, and procedures will adequately protect public health and safety. The staff evaluated the applicant's existing facility designs and procedures, which are in various stages of completion, together with the applicant's stated commitments to complete certain design and procedures according to criteria specified by the applicant. If the Commission issues a license, it may contain conditions and limitations imposed to assure compliance with applicable regulations. The determination whether there is reasonable assurance that public health and safety will be adequately protected will be based in part on a comparison of the regulatory requirements against existing information, applicant commitments, and conditions imposed by the staff.

Once a license is granted (assuming the Agency decision is to issue a license), construction of the facility may begin. In accordance with 10 CFR 70.72(d)(2), the applicant (then licensee) will submit to the NRC annual updates to the Integrated Safety Analysis Summary along with a brief summary of the changes made during the year. In addition, the applicant has committed to provide to the NRC an update to the ISA Summary at least 180-days prior to the planned introduction of special nuclear material into the ACP facility. The NRC will review these submissions as well as any license amendment requests that may be submitted.

Although the applicant (then licensee) can start construction following issuance of the license, it may not begin operation of the enrichment facility until after it successfully completes a second step. Prior to operation, the Commission must verify through inspection that the facility has been constructed in accordance with the requirements of the license (see 10 CFR 70.32(k)). Only after this step is successfully completed will the enrichment facility be allowed to begin operations.

The staff used several guidance documents to evaluate the applicant's license application (LA)

and to complete the Safety Evaluation Report (SER) for the American Centrifuge Plant (ACP). The staff primarily used NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." In addition, for the staff's review of the safeguards section of the LA, the staff used NUREG-1065, "Acceptable Standard Format and Content for the Fundamental Nuclear Material Control Plan Required for Low-Enriched Uranium Facilities," for its review of the security related portions the staff used Regulatory Guide 5.67, "Material Control and Accounting for Uranium Enrichment Facilities Authorized To Produce Special Nuclear Material of Low Strategic Significance," and, for its review of the physical protection, the staff used 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."

NUREG-1520 was the primary document used to assist in defining the scope, level of detail, and acceptance criteria for reviews. NUREG-1520 provides generic guidance for reviewing and evaluating the health, safety, and environmental protection aspects of applications for licenses to possess and use special nuclear material in nuclear fuel cycle facilities. The principal purpose of NUREG-1520 is to ensure the quality and uniformity of reviews conducted by the staff. Because NUREG-1520 describes the scope, level of detail, and acceptance criteria for reviews, it also serves as regulatory guidance for applicants who need to determine what information to present in a LA. The information presented in NUREG-1520 does not preclude licensees or applicants from identifying portions of NUREG-1520 that are not directly applicable or from suggesting alternative approaches to those specified in NUREG-1520 to demonstrate compliance with applicable regulations because NUREG-1520 is a guidance document. Should a licensee or applicant suggest alternative approaches, the staff retains the responsibility to make an independent determination concerning the adequacy of the applicant's proposed approaches.

NUREG-1520 was developed as a generic document for licensing fuel cycle facilities under 10 CFR Part 70, including fuel fabrication facilities and uranium enrichment facilities. Extensive communication occurred with current fuel cycle licensees to ensure that all necessary safety and environmental issues were addressed. While it is true that there are differences among these types of plants and among the relative risks of certain hazards at different fuel cycle facilities, hazards that will exist at the proposed ACP are similar to the types of hazards at other fuel cycle facilities for which NUREG-1520 was prepared. These hazards include handling of uranium hexafluoride (UF_6) cylinders, processing of UF_6 as a gas and sometimes as a liquid, use of autoclaves or similar devices for feeding and sampling uranium, nuclear criticality, and equipment decontamination operations.

The staff adapts the generic standard review plan to review applications for different types of 10 CFR Part 70 facilities. The relative risk of the facility depends on the specific hazards associated with a particular technology (e.g., enrichment facility, fuel fabrication facility, or mixed-oxide [MOX] fuel fabrication facility). The staff's review of each type of LA would focus on those specific types of hazards. The goal of the reviews is to determine whether applicable regulatory requirements are met to ensure that an adequate level of safety is provided to protect the health and safety of the public and the environment. Specific regulatory requirements for each type of license are found in the applicable sections of NRC's regulations. The staff recognizes that the types and magnitudes of potential hazards varies greatly between the various types of licensees and even within each type of hazard. Based on the processes performed at each type of facility, overall, the proposed ACP facility has the lowest level of potential hazard, fuel fabrication facilities have the next level of hazard, and the MOX fuel

fabrication facility has the highest level of hazard of all 10 CFR Part 70 fuel cycle facilities. Thus, while the guidance in NUREG-1520 is generally applicable to a gas centrifuge uranium enrichment facility, with a few exceptions, the staff's review is informed by the fact that the overall risk of this type of facility is lower than that of other types of fuel facilities licensed by NRC.

The applicant also submitted an Environmental Report, which was used to prepare NUREG-1834, "Final Environmental Impact Statement for the Proposed American Centrifuge Plant in Piketon, Ohio," which was made available to the public in April 2006.

A summary of NRC's review and findings in each of the review areas is provided below:

General Information

The applicant provided an adequate description of the facility and processes so that the staff has an overall understanding of the relationships of the facility features as well as the function of each feature. Financial qualifications were properly explained and outlined in the application. The description of the site included important information about regional hydrology, geology, meteorology, the nearby population, and potential effects of natural phenomena at the facility. The applicant has requested an exemption from the requirement to purchase liability insurance.

Organization and Administration

The applicant adequately described the responsibilities and associated resources for the design, construction, and operation of the facility and its plans for managing the project. The plans and commitments described in the application provide reasonable assurance that an acceptable organization, administrative policies, and sufficient competent resources have been established or committed for the design, construction, and safe operation of the facility.

"Integrated Safety Analysis" (ISA) and ISA Summary

The applicant provided sufficient information about the site, facility processes, hazards, and types of accident sequences. The information provided addressed each credible event, the potential radiological and chemical consequences of the event, and the likelihood of the event. No mitigated event consequence exceeds the performance requirements of 10 CFR 70.61. The applicant also provided adequate information about items relied on for safety (IROFS). A license condition will be added to the license, if issued, to ensure that IROFS boundaries will be defined using the applicant's IROFS boundary definition procedure.

Radiation Protection

The applicant provided sufficient information to evaluate the Radiation Protection Program. The application adequately describes: (a) the qualification requirements; (b) written radiation protection procedures; (c) the radiation work permit program; (d) the program for ensuring that worker and public doses are as low as reasonably achievable (ALARA); and (e) necessary training for all personnel who have access to radiologically restricted areas. The radiation survey and monitoring program is adequate to protect workers and members of the public who may be potentially exposed to radiation.

The applicant has requested two exemptions, one regarding the labeling of UF_6 feed, product, and depleted uranium cylinders, and one requiring the labeling of containers located in the Restricted Areas within the ACP. The applicant has also requested approval for an alternative method for controlling access to high radiation areas. In addition, a special authorization request was made regarding surface contamination levels for the release of items for unrestricted use.

Nuclear Criticality Safety

The applicant provided adequate information to evaluate the Nuclear Criticality Safety (NCS) program. The applicant committed to having an adequate group of qualified staff to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures. The program meets the regulatory requirements. The applicant requested an exemption from criticality monitoring of the UF_6 cylinder storage yards.

Chemical Process Safety

The applicant adequately described and assessed accident consequences that could result from the handling, storage, or processing of licensed materials and that could have potentially significant chemical consequences and effects. The applicant performed hazard analyses that identified and evaluated chemical process hazards and potential accidents and established safety controls that meet the regulatory requirements.

Fire Safety

The applicant committed to reasonable engineered and administrative controls to minimize the risk of fires and explosions. The IROFS and defense-in-depth protection discussed in the applicant's ISA Summary, along with safety basis assumptions and the planned programmatic commitments in the license application, meet safety requirements and provide reasonable assurance that the facility is protected against fire hazards.

Emergency Management

The applicant provided an adequate Emergency Plan for the facility that meets the regulatory requirements. The applicant commits to maintaining and executing an Emergency Plan for responding to the radiological and chemical hazards resulting from potential release of radioactive or chemically hazardous materials incident to the processing of licensed material. The requirements of the Emergency Plan are implemented through approved written procedures.

Environmental Protection

The applicant committed to adequate environmental protection measures, including: (a) environmental and effluent monitoring; and (b) effluent controls to maintain public doses ALARA as part of the radiation protection program. The applicant's proposed controls are adequate to protect the environment and the health and safety of the public and comply with the regulatory requirements.

Decommissioning

The applicant provided a conceptual decommissioning plan for the facility that addresses: (a) contamination control; (b) control of worker exposures and waste volumes; (c) waste disposal; (d) the final radiation survey; (e) control of SNM; (f) control of classified matter; and (g) record-keeping for decommissioning.

The applicant provided a decommissioning funding plan for the facility that demonstrates that adequate funding will be available for decommissioning and that decommissioning will not pose a threat to public health and safety or the environment. The applicant also submitted an exemption request to allow for incremental funding for depleted uranium disposition based on the expected number of centrifuges to be built and installed and on the expected amount of depleted uranium tails to be generated annually in a forward-looking manner. The decommissioning funding plan and the incremental approach for funding depleted uranium disposition costs will provide adequate assurance for decommissioning funding because sufficient funding will be available to decommission the facility and disposition the inventory of depleted uranium on-site at any point in time. The applicant intends to provide for decommissioning funding through a surety bond. However, the applicant may choose alternate funding methods prepared in accordance with the requirements of 10 CFR Part 70 and guidance provided in NUREG 1757, Volume 3, Appendix A.

The applicant also provided proposed language for a surety bond which will be executed if an alternate funding method is not chosen before the applicant commences operations and takes possession of licensed material. The applicant will update the site-specific cost estimate at least every 3 years, to reflect inflation and changes in site inventories and conditions, that could affect the cost of decommissioning. A license condition will be added to the license, if issued, to ensure that the applicant provides final copies of the proposed financial assurance instruments to NRC for review at least six months prior to the planned date for possessing licensed material, and provide to NRC final executed copies of the reviewed financial assurance instruments prior to the receipt of licensed material. The decommissioning funding plan is acceptable because it provides sufficient funding to ensure decommissioning and decontamination of the facility can be accomplished even if the licensee is unable to meet its financial obligations.

Management Measures

The applicant provided information about management measures that will be applied to the project. The information describes: (a) the overall configuration management program and policy; (b) the maintenance program; (c) training; and (d) the process for the development, approval, and implementation of procedures. The applicant explained the audits and assessments program as well as incident investigations and records management system. The applicant committed to establishing and documenting surveillances, tests, and inspections to provide reasonable assurance of satisfactory performance of the IROFS. The proposed management measures are acceptable and meet the regulatory requirements in 10 CFR 70.62(d). The applicant requested an exemption from the reporting criteria for issuing a written follow-up report within 30 days of the initial event report.

Materials Control and Accountability

The applicant provided information describing the FNMCP for the project. The FNMCP describes the programs to be used to control and account for SNM in the facility. The program meets the applicable regulatory requirements in Part 74.

Physical Protection

The applicant provided information on the policies, methods, and procedures to be implemented to protect SNM of low strategic significance used and possessed at the facility. This information is acceptable and meets the requirements in Part 73.

The applicant also provided information on the protection of classified matter, including security controls and procedures, to ensure that classified matter is used, processed, stored, reproduced, transmitted, transported, and destroyed appropriately. This program is acceptable and in accordance with the regulatory requirements in Part 95 for a facility clearance.

Transportation Security

The applicant provided information in the Physical Security Plan on the policies, methods, and procedures to be implemented to protect SNM of low strategic significance in transit to and from the facility. This information is acceptable and meets the requirements in Part 73.

LIST OF ACRONYMS AND ABBREVIATIONS

ACL	Administrative control level
ACP	American Centrifuge Plant
ACR	Area Control Room
AIHA	American Industrial Hygiene Association
ALARA	As Low as Reasonably Achievable
ALOHA	Area Locations of Hazardous Atmospheres
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOA	Area of Applicability
ARF	Airborne Release Fraction
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BDC	Baseline Design Criteria
BEQ	Baseline Effluent Quantity
CA	Contamination Area
CAA	Controlled Access Area
CAAS	Criticality Accident Alarm System
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
Ci	Curies
cm	Centimeter
CM	Configuration Management
CMP	Classified Matter Plan
CRR	Cyclic Resistance Ratio
CSR	Cyclic Stress Ratio
DA	Design Authority
DAC	Derived Air Concentration
DBE	Design Basis Earthquake
DCP	Double Contingency Principle
DFP	Decommissioning Funding Plan
DOE	United States Department of Energy
DOT	United States Department of Transportation
DR	Damage Ratio
DUF_6	Depleted Uranium Hexafluoride
EA	Environmental Assessment
EAL	Emergency Action Levels
ECS	ECS, LLP
EOC	Emergency Operations Center
EP	Emergency Plan

EPA	United States Environmental Protection Agency
EPRI	Electrical Power Research Institute
ER	Environmental Report
ERO	Emergency Response Organization
ERPG	Emergency Response Planning Guidelines
EV	Evacuation Vacuum
FHA	Fire Hazards Analysis
FLM	Front Line Managers
FMO	Fissile Material Operations
FNMCP	Fundamental Nuclear Material Control Plan
FOCI	Foreign Ownership, Control, or Influence
ft	Feet
g	Acceleration of Gravity
GCEP	Gas Centrifuge Enrichment Plant
GDP	Gaseous Diffusion Plant
gpm	Gallons per Minute
HEPA	High Efficiency Particulate Air
HF	Hydrogen Fluoride
HP	Health Physics
HRA	High Radiation Area
HS&E	Health, Safety, and Environment
IEEE	Institute for Electrical and Electronics Engineering
IHS	Industrial Hygiene and Safety
in	Inch
IROFS	Items Relied on for Safety
ISA	Integrated Safety Analysis
ISG	Interim Staff Guidance
ISTP	Integrated Systems and Test Plan
I&C	Instrumentation and Control
k_{eff}	K effective
kg	Kilogram
km	Kilometer
kPa	KiloPascals
kPa/s	KiloPascals per Second
L	Liter
LA	License Application
lb	pound
LCF	Latent Cancer Fatalities
LEC	Liquid Effluent Collection
LLNL	Lawrence Livermore National Laboratory
LPF	Leak Path Factor
LSS	Low Strategic Significance

m	Meter
m^3	Cubic Meter
MAR	Material at Risk
MC&A	Material Control and Accounting
MCW	Machine Cooling Water
µCi	Microcuries
MDC	Minimum Detectable Concentration
MEI	Maximally Exposed Individual
M&TE	Measuring and Test Equipment
MOU	Memorandum of Understanding
MOX	Mixed Oxide
MREM	Millirem
MSDS	Material Safety Data Sheet
mg	Milligram
mi	Mile
µg	Microgram
mm	Millimeter
m/s	Meter per Second
MT	Metric Ton
NCS	Nuclear Criticality Safety
NCSE	Nuclear Criticality Safety Evaluation
NEPA	National Environmental Protection Act
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NMC&A	Nuclear Materials Control and Accountability
NMMSS	Nuclear Materials Management and Safeguards System
NPH	Natural Phenomena Hazard
NPDES	National Pollutant Discharge Elimination System
NOAA	National Oceanic and Atmospheric Administration
NRC	United States Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
OJT	On-the-job Training
OSHA	United States Occupational Safety and Health Administration
PA	Public Address
PAG	Protective Action Guidelines
PFPE	Perfluorinated Polyether
PHA WI/CL	Preliminary Hazard Analysis and What if/Checklist
PM	Preventive Maintenance
PMF	Probably Maximum Flood
PORTS	Portsmouth Gaseous Diffusion Plant
psf	Pounds per Square Foot
psi	Pounds per Square Inch
PSM	Process Safety Management
PSRC	Plant Safety Review Committee
PSP	Physical Security Plan

PSS	Plant Shift Superintendent
PV	Purge Vacuum
QA	Quality Assurance
QAPD	Quality Assurance Program Description
QL	Quality Level
RAI	Request for Additional Information
RCW	Recirculating Cooling Water
REIRS	Radiation Exposure Information Reporting System
rem	Roentgen Equivalent Man
RF	Respirable Fraction
RMDC	Records Management and Document Control
RP	Radiation Protection
RPM	Radiation Protection Manager
RWP	Radiation Work Permit
SAE	Site Area Emergency
SEC	U.S. Securities and Exchange Commission
SEI	Structural Engineering Institute
SER	Safety Evaluation Report
SNM	Special Nuclear Material
SNM-LSS	Special Nuclear Material - Low Strategic Significance
SSC	Structures, Systems, and Components
STP	Sewage Treatment Plant
Sv	Sievert
SWU	Separative Work Unit
Tc-99	Technetium-99
TDAG	Training Development and Administrative Guide
TEDE	Total Effective Dose Equivalent
TLV	Threshold Limiting Value
TWC	Tower Water Cooling
UF_4	Uranium Tetrafluoride
UF_6	Uranium Hexafluoride
UO_2	Uranium Dioxide
UO_2F_2	Uranyl Fluoride
USEC	USEC Inc.
USGS	United States Geological Survey
^{235}U, U-235	Uranium-235
wt	Weight
WI/CL	What If/Checklist

1.0 GENERAL INFORMATION

1.1 FACILITY AND PROCESS DESCRIPTION

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's facility and process description is to evaluate whether the application includes an overview of the facility layout and a summary description of the proposed processes. A more detailed description of the facility and processes is contained in the "Integrated Safety Analysis (ISA) Summary" (USEC, 2006b).

1.1.1 REGULATORY REQUIREMENTS

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 the applicant will use 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.

1.1.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the facility and process description section of the license application (LA) (USEC, 2006c) is contained in Chapter 1 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). This chapter is applicable in its entirety. For information regarding exemption requests, see Section 1.2.3.6, "Special Exemptions or Special Authorizations," of this chapter. The acceptance criteria applicable to this review are contained in Section 1.1.4.3 of NUREG-1520 (NRC, 2002).

1.1.3 STAFF REVIEW AND ANALYSIS

In Section 1.1 of the LA (USEC, 2006c), the applicant provides a summary description of the proposed gas centrifuge uranium enrichment plant and processes. The applicant is proposing to use a gas centrifuge enrichment process to enrich uranium. The proposed plant, known as the American Centrifuge Plant (ACP), will have a nominal production capacity of 3.5 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 (UF_6). Gaseous UF_6 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, uranium-235 (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 applicant has proposed that the ACP be located at the U.S. Department of Energy's (DOE's) Portsmouth Gaseous Diffusion Plant (PORTS) reservation in Piketon, Ohio, in refurbished existing buildings, newly constructed facilities, and grounds to be leased from DOE by the United States Enrichment Corporation, a subsidiary of the applicant. The applicant will in turn, sub-lease the ACP buildings and grounds from the United States Enrichment Corporation.

NRC will not issue a license until the lease is signed and NRC confirms that the contents of the lease agreements do not contradict any license conditions or considerations and assumptions documented in the applicant's LA (USEC, 2006c) and its supporting documents and NRC's licensing basis as documented in this Safety Evaluation Report (SER), Final Environmental Impact Statement, and their supporting documents.

The ACP facility will consist of multiple buildings, each one of which will performs a specific function. A listing of selected buildings and their specific functions follows:

1. Two existing Process Buildings, X-3001 and X-3002, which will be refurbished to: (a) house operating centrifuge machines; (b) associated process piping; (c) instrumentation and controls; (d) computer systems; and (e) auxiliary support equipment. The X-3001 and X-3002 buildings will be similar in construction, layout, and design.

2. The facilities will include the existing X-3012 Process Support Building to provide operational control and maintenance of the equipment in the Process Buildings.

3. The existing X-3346 Feed and Customer Services Building will provide for process feed, sampling, and product transfer requirements. The Feed Area of the building will house electrically heated feed ovens to provide the UF_6 feed. UF_6 feed will be processed through purification burp systems before being fed into the process piping in X-3001 and X-3002. A bridge crane will be used to place feed cylinders on railcarts.

4. The Customer Services Area of the X-3346 building will house the back-end equipment necessary for sampling and transfer of UF_6 material to customer cylinders. The Customer Services Area will be the only area where liquid UF_6 may be present. In this building, UF_6 product contained in 10-ton cylinders will be liquified in electrically heated containment autoclaves for the purpose of sampling and transferring the UF_6 into 2.5 ton customer cylinders.

5. The new X-3346A Feed and Product Shipping and Receiving Building will serve as the focal point for the receipt and shipping of natural and enriched uranium in U.S. Department of Transportation approved containers.

6. The new X-3356 Product and Tails Withdrawal Building will house the equipment for withdrawal of the enriched and depleted UF_6 from the X-3001 and X-3002 Process Buildings. In this building, UF_6 product will be desublimed into cold traps before transfer into 10-ton product cylinders via sublimation in the cold traps, followed by desublimation in the cylinders. Tails withdrawal will be performed via compression and direct desublimation of the UF_6 into 14-ton tail cylinders.

7. The X-7725 Recycle/Assembly Facility will provide an area where centrifuge machines can be manufactured, assembled, tested, and maintained.

8. The X-7726 Centrifuge Training and Test Facility will provide areas to receive and test centrifuge components, and to assemble and repair the centrifuges. The facility may also be used as a machine assembly training area for the ACP.

9. The X-7727H Interplant Transfer Corridor will provide a protected pathway for

transporting centrifuge machines between the X-7725 or X-7726 buildings and the Process Buildings.

10. Cylinder Storage Yards will support the movement and storage of cylinders containing UF_6 material.

11. The X-2232C Interconnecting Process Piping connects the X-3346 Building to the X-3001 and X-3002 Process Buildings and will be external to the primary facilities.

12. Secondary facilities for the ACP will include data processing facilities, emergency response facilities, electrical distribution systems, security fencing and portals, a pumphouse, an air generation plant, a cooling tower, a boiler system, a training facility, a maintenance facility, storage facilities, and waste accountability facilities.

The LA (USEC, 2006c) provides additional descriptions and process details, including drawings, for each of these areas.

Potential waste streams that will be generated at the ACP will include low-level radioactive waste, low-level mixed waste, hazardous waste, sanitary/industrial waste, recyclable waste, and classified/sensitive waste. Effluent discharge points are discussed in Chapter 9 of this SER. Depleted UF_6 tails will be stored in steel cylinders, within cylinder storage yards, until the cylinders are transferred to DOE or another facility for deconversion; until decommissioning; or until they are transferred to another licensee for commercial reuse. At or before the time of decommissioning, any remaining UF_6 tails will be converted to a stable form and disposed of in accordance with the USEC Privatization Act and other statutory authorizations and requirements at DOE's UF_6 conversion facility or another licensed facility.

As stated above, the applicant provided information at a level of detail that is appropriate for general familiarization and understanding of the proposed facility and processes. The application summarizes the facility information contained in the ISA Summary and includes descriptions of the overall facility layout on scaled drawings, including the site's geographical features and facility structural features and transportation rights-of-way. The summary also describes the relationship of specific facility features to the major processes that will be ongoing at the facility. The major chemical and mechanical processes involving licensable material are described in summary form, based in part on information presented in the ISA Summary. This description includes: (a) reference to the building locations of major process components; (b) brief descriptions of the process steps; (c) the chemical forms of licensable material in process; and (d) the types, amounts and discharge points of waste materials discharged to the environment from the processes. The applicant presented a summary identification of the raw materials, by-products, waste, and finished products of the facility. This information included data about expected levels of trace impurities or contaminants characterized by identity and concentration. The information the applicant provided meets the guidance in Section 1.1.4.3(1), (2), (3), and (4) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.1.4 EVALUATION FINDINGS

The staff has reviewed the proposed general facility and process descriptions according to Section 1.1 of NUREG-1520 (NRC, 2002). The applicant has adequately described: (1) the facility and processes so that the staff has an overall understanding of the relationships of the

facility features; and (2) the function of each feature. The staff concludes that the applicant has met the requirements and acceptance criteria applicable to this section.

1.2 INSTITUTIONAL INFORMATION

The purpose of NRC's review of the applicant's institutional information is to evaluate whether the application includes adequate information identifying the applicant, the applicant's characteristics, and the proposed activity.

1.2.1 REGULATORY REQUIREMENTS

The regulations in 10 CFR 30.32 and 10 CFR 40.31 require each application for a license to include: (a) information on the identity of the applicant; (b) name, chemical and physical form, and maximum amount that will be possessed; and (c) purpose for which the licensed material will be used. The regulations in 10 CFR 70.22 require each application for a license to include: (a) information on the corporation applying for a license; (b) the location of the principal office; (c) the names and citizenship of the principal officers; (d) information concerning ownership and control; (e) the proposed site activities; (f) financial qualifications; and (g) the name, amount, and specifications of the licensed material to be used. The regulations in 10 CFR 70.23(a)(5) require that the applicant appears to be financially qualified to engage in the proposed activities in accordance with the regulations. The regulations in 10 CFR 40.38 and 10 CFR 70.40 place restrictions on the ownership of USEC. 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.

1.2.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The acceptance criteria applicable to NRC's review of the institutional information section of the application are contained in 10 CFR 30.32, 10 CFR 40.31, 10 CFR 40.38, 10 CFR 70.23(a)(5), 10 CFR 70.40, 10 CFR Part 95, 10 CFR 140.13b, and Section 1.2.4.3 of NUREG-1520 (NRC, 2002). Chapter 1 of NUREG-1520 is applicable to the ACP facility in its entirety. Section 1.2.3.6, "Special Exemptions or Special Authorizations," of this chapter, addresses exemptions and special authorizations.

1.2.3 STAFF REVIEW AND ANALYSIS

1.2.3.1 Corporate Identity

In Section 1.2.1 of the LA (USEC, 2006c), the applicant, USEC, provided information on the corporate organization. USEC, including its wholly owned subsidiaries, was organized under Delaware law in connection with the privatization of the United States Enrichment Corporation. Section 1.2.1 of the LA (USEC, 2006c) identifies the principal officers of the applicant and provides information on the locations of the applicant's principal office and the ACP facility.

United States Enrichment Corporation, a wholly owned subsidiary of USEC, operates the Paducah and Portsmouth gaseous diffusion plants (GDPs). In 1997, NRC issued Certificates of Compliance to the United States Enrichment Corporation to operate the Paducah and Portsmouth GDPs (Docket Numbers 70-7001 and 70-7002, respectively). NRC assumed regulatory oversight for these operations from DOE on March 3, 1998. In February 2004, NRC

issued a license to USEC to operate the Lead Cascade Demonstration Facility (Docket No. 70-7003) pursuant to 10 CFR Part 70.

In November 2004, USEC acquired NAC International from Pinnacle West Capital Corporation. In addition to developing radioactive material transportation casks and dry cask storage systems for spent nuclear reactor fuel, NAC International operates the Nuclear Materials Management and Safeguards System (NMMSS) for DOE and NRC. NMMSS is a national database system that accounts for and tracks special nuclear material (SNM).

The USEC principal office is located at 6903 Rockledge Drive, Bethesda, MD 20817. USEC is listed on the New York Stock Exchange under the ticker symbol USU. Private and institutional investors own the outstanding shares of USEC. The mailing address for the ACP is: USEC Inc., American Centrifuge Plant, P.O. Box 628, Piketon, Ohio 45661.

As stated above, the applicant furnished its full name and address, a full description of the proposed facility site location, the State where the corporation is incorporated, and the location of the principal offices. The applicant also described the corporation's control and ownership, including any control or ownership exercised by a foreign entity. The applicant provided information on primary ownership, and relationships to other components of the organization of the same ownership. The applicant described the presence and operations of any other organization on the site. The information the applicant provided meets the guidance in Section 1.2.4.3(1) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.2.3.2 Foreign Ownership, Control, or Influence (FOCI)

As part of the process of transitioning regulatory oversight of the United States Enrichment Corporation's operation of the two GDPs from DOE to NRC, NRC had requested DOE to conduct a FOCI analysis for the United States Enrichment Corporation. Based on DOE's analysis, on August 18, 1998, NRC rendered the United States Enrichment Corporation and its parent company, USEC, a positive FOCI determination and ultimately a facility clearance at the SECRET-Restricted Data level. A positive FOCI determination indicates that the degree and extent of FOCI over the United States Enrichment Corporation and USEC does not pose an undue risk to the national security. It also indicates that the United States Enrichment Corporation and USEC are not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. Therefore, issuance of a license would not be inimical to the common defense and security of the United States or to the maintenance of a reliable and economical domestic source of enrichment services and a license can be issued under the requirements of 10 CFR 40.38 and 10 CFR 70.40.

On October 5, 2005, the United States Enrichment Corporation again submitted FOCI material to DOE so DOE could perform a FOCI analysis for NRC, which will ultimately render a FOCI redetermination for the United States Enrichment Corporation and USEC. National initiatives require that a FOCI redetermination be conducted approximately every 5 years. On June 15, 2006, DOE transmitted a letter (DOE, 2006) to USEC documenting that a favorable determination had been made. NRC accepts DOE's FOCI analyses based on an Interagency Agreement between NRC and DOE dated May 6, 2002 (DOE, 2002).

1.2.3.3 Financial Qualifications

1.2.3.3.1 Project Costs

Section 1.2.2 of the LA (USEC, 2006c) states that the estimated project cost to build and operate the ACP at initial SWU capacity would approach $1.5 billion in 2004 dollars, excluding capitalized interest, tails disposition, decommissioning, and any replacement equipment required during the life of the plant outside of normal spare equipment. Since the ACP design is modular, the applicant plans to construct the plant and install machines in phases until it reaches the initial capacity of 3.5 million SWU, approximately 4 years after receipt of a license. The LA (USEC, 2006c) states that, after NRC approval of applicable license amendments, the applicant may construct and install additional capacity thereafter, as operations and market conditions permit.

In Appendix C to the LA (USEC, 2006c), the applicant provided an estimate of the construction and startup costs. Projected costs were summarized for process machinery and equipment; building and infrastructure; centrifuge machines; engineering, procurement, and construction; manufacturing equipment; regulatory compliance; and other costs. Appendix C also included spreadsheets showing these costs and costs associated with major buildings, equipment, and labor activities as incurred by year from 2005 to 2010.

The staff considers the construction cost estimate, as presented in the LA (USEC, 2006c), and in supporting supplements in response to the staff's request for additional information, reasonable and, therefore, the staff concludes that the $1.5 billion estimate (USEC, 2006c) is a reasonable estimate for constructing the facility.

1.2.3.3.2 Financial Qualifications

The applicant anticipates that its funding for various phases of construction may come from funds from operations, capital raised by itself, potential partners, and lending and/or lease arrangements. Before initiating each phase of construction, the applicant will make its cost estimate for such a phase available, and document to NRC the source of funds available or committed to fund that phase; these conditions will be made part of the license.

The applicant is a publicly held global energy company, thus it is required to submit Form 10-K Annual Report (10-K) to the Securities and Exchange Commission (SEC, 2002, and 2003). The LA (USEC, 2006c) included, and the staff reviewed, the applicant's 2003 10-K, which includes selected financial data for 1999 to 2003, and more detailed data for 2001 to 2003 in its consolidated financial statements. A report of the Independent Auditors is also provided. The Independent Auditor concluded that the applicant's financial statements "present fairly, in all material respects, the financial position of the applicant and its subsidiaries at December 31, 2003 and 2002, and the results of its operations and cash flows for the year ended December 31, 2003, the six-month period ended December 31, 2002, and the fiscal year ended June 30, 2002, in conformity with accounting principles generally accepted in the United States of America."

The staff also reviewed the applicant's 2005 10-K (SEC, 2005), which includes selected financial data through December 31, 2005. An independent registered public accounting firm audited the applicant's 2004 and 2005 consolidated financial statements and certified that they were in conformity with generally accepted accounting principles.

The staff's review of the above-referenced 10-Ks revealed that, for the year ending December 31, 2005, the applicant had total assets of over $2 billion, total revenue of nearly $1.6 billion, net income of $22.3 million, and cash and cash equivalents of $259 million. Total assets have been about $2 billion since 2002. Total revenue in 2004, 2003, and 2002 was approximately $1.4 billion per year. Net income was $23.4 million in 2004, $9.0 million in 2003, and a loss of $3.3 million in 2002. Cash and cash equivalents were between $171 and $214 million for the years 2002-2004.

To ensure the applicant meets the financial qualifications requirements for construction and operation of the facility, the staff is imposing the following license conditions:

> *Construction of each incremental phase of the ACP shall not commence before funding for that increment is available or committed. Of this funding, USEC Inc. must have in place before constructing such increment, commitments for one or more of the following: equity contributions from USEC Inc., affiliates and/or partners, along with lending and/or lease arrangements that solely or cumulatively are sufficient to ensure funding for the particular increment's construction costs. USEC Inc. shall make available for NRC inspection, documentation of both the budgeted costs for such phase and the source of funds available or committed to pay those costs.*
>
> *Operation of the ACP shall not commence until USEC Inc. has in place either: (1) long term contracts lasting five years or more that provide sufficient funding for the estimated cost of operating the facility for the five year period; (2) documentation of the availability of one or more alternative sources of funds that provide sufficient funding for the estimated cost of operating the facility for five years; or (3) some combination of (1) and (2).*

Therefore, based on the applicant having a reasonable construction cost estimate of $1.5 billion, as discussed in Section 1.2.3.3.1 of this SER, and a reasonable approach for financing the construction and operation of the facility, the staff finds the applicant has reasonable assurance of securing the necessary financial resources needed for constructing and operating the ACP project. Therefore, the staff finds that the applicant appears to be financially qualified to build and operate the proposed ACP, in accordance with Section 1.2.4.3(2) of NUREG-1520 (NRC, 2002), 10 CFR 70.22(a)(8), and 10 CFR 70.23(a)(5).

1.2.3.3.3 Liability Insurance

Under 10 CFR 140.13b, a uranium enrichment facility is required to carry 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.

According to DOE's lease agreement for the PORTS, the United States Enrichment Corporation, a subsidiary of the applicant, is indemnified under Section 170d of the *Atomic Energy Act* for liability claims arising out of any occurrence within the United States, causing, within or outside the United States, bodily injury, sickness, disease, or death, or loss of or damage to property, or loss of use of property, arising out of or resulting from the radioactive,

toxic, explosive, or other hazardous properties of chemical compounds containing source or SNM arising out of activities under the lease. The applicant provided a letter from DOE, dated March 7, 2005 (DOE, 2005), that provided draft language from the lease agreement in which DOE will indemnify the ACP. According to the applicant, DOE will provide indemnity for the ACP as it had done for the Portsmouth and Paducah GDPs unless liability insurance is commercially available at commercially reasonable rates. Therefore, the applicant has requested an exemption to 10 CFR 40.31(l) and 70.22(n) to use the DOE indemnity in lieu of providing liability insurance if it is unable to obtain commercially reasonable rates provide commercial liability insurance. Section 1.2.3.6 of this SER discusses the applicant's exemption request. Such an indemnification provided by DOE to the applicant for the ACP will be sufficient to meet the requirements of Section 193(d) of the *Atomic Energy Act* of 1954, as amended, and 10 CFR 140.13b. NRC staff considers DOE's indemnification of the ACP as adequate means to demonstrating compliance with the requirements of 10 CFR 140.13b.

To allow NRC to confirm DOE's indemnification of the ACP or proof of liability insurance, the staff is imposing the following condition:

> *USEC Inc. shall provide to the Commission, at least 120 days prior to the planned date for obtaining licensed material, documentation of any liability insurance required to be obtained by USEC Inc. under its lease with DOE for the ACP by that time or, alternatively, the status of USEC Inc.'s efforts to obtain any such liability insurance. During the time that USEC Inc. is engaged in efforts to obtain liability insurance, USEC Inc. shall provide the Commission with status reports regarding those efforts. The status reports shall be submitted at a frequency of at least once every six months following issuance of a license. USEC Inc. shall notify the Commission within 30 days upon receiving notification of denial or approval of commercial liability insurance for the ACP. If commercial liability insurance is required to be obtained under its lease with DOE, within 60 days of receiving notification of approval of commercial liability insurance, USEC Inc. shall provide proof of liability insurance coverage and a justification, for Commission review and approval, if USEC Inc. is proposing to provide less than $300 million of liability insurance coverage.*

During the 120 days, NRC staff will confirm DOE's indemnification of the ACP or proof of liability insurance.

1.2.3.3.4 American Centrifuge Lead Cascade Facility

In February, 2003, the applicant submitted to NRC, its LA (USEC, 2003c) and environmental report (ER) for the American Centrifuge Lead Cascade Facility (Lead Cascade) (USEC, 2003a). The Lead Cascade, a demonstration and test facility, will have up to 240 centrifuges. These will be housed in the X-3001 Process Building of the ACP. The only enriched uranium that the applicant will withdraw from the Lead Cascade will be in the form of small samples.

NRC completed its review of the Lead Cascade application and in January 2004, issued its Environmental Assessment with a "Finding of No Significant Impact" and SER. On February 17, 2004, DOE leased the Lead Cascade facility to the United States Enrichment Corporation, which subsequently subleased the facility to the applicant. After determining that the lease allowed the applicant to refurbish and subsequently operate the facility in accordance with its

Lead Cascade LA (USEC, 2003c), NRC issued a 5-year possession and use Materials License (SNM-7003) to the applicant on February 24, 2004. During operations, the Lead Cascade facility will employ about 50 workers.

The applicant states in Section 1.1.8 of the LA (USEC, 2006c) that it plans to transition activities from the Lead Cascade possession and use license to the ACP construction and operations license. In Section 1.1.8 of the LA (USEC, 2006c), the applicant presented four options for transitioning the Lead Cascade to the ACP. These options include:

1. Subsume Lead Cascade operations under the ACP;

2. Renew the Lead Cascade license;

3. Terminate the Lead Cascade operations; and

4. Have phased deployment.

For options 1 and 4, where Lead Cascade operations transition to ACP operations, the options include the submittal for NRC review and approval under 10 CFR 40.42 and 70.38 of a Lead Cascade license termination plan and plan to safely transition operations to the ACP license. ACP operations involving licensed material will not take place in the Lead Cascade until the Lead Cascade license is terminated. The staff reviewed these general transition options, and found that sufficient management control is proposed to enable the transitions to occur in accordance with NRC regulations in 10 CFR Parts 40 and 70.

1.2.3.3.5 NRC/DOE Memorandum of Understanding

On March 24, 2004, NRC and DOE signed a Memorandum of Understanding (MOU) (NRC, 2004a) which delineates each Agency's roles and responsibilities concerning regulatory oversight of the Lead Cascade to avoid dual regulation. The MOU will also ensure that the applicant's activities, to be conducted during centrifuge disassembly/assembly and operational phases of the Lead Cascade, have regulatory continuity and therefore, are safe and secure. According to the MOU, from the time of issuance of an SNM and Source Material possession license on February 24, 2004, until the time UF_6 is introduced in the Lead Cascade, NRC's oversight responsibility would be limited to determining the adequacy of the management measures, including quality control applied to items relied on for safety, and other USEC activities addressed by NRC safety and safeguards requirements as documented in USEC's Lead Cascade application. During this time, DOE would provide regulatory oversight for safeguards and security and any existing and as-found safety issues. According to the MOU, NRC will assume regulatory jurisdiction from DOE for the Lead Cascade at the time UF_6 is first introduced to the facility. However, DOE will retain its jurisdiction for providing personnel access authorization.

A separate DOE/NRC MOU will be developed and issued for the ACP before any license for this facility is issued. The MOU for the ACP is expected to be similar to that for the Lead Cascade.

1.2.3.4 Type, Quantity, and Form of Licensed Material

Table 1.2-1 of the LA (USEC, 2006c) lists the type, quantity, and form of the licensed material

proposed for possession. The applicant proposes to use and possess up to 4,000 metric tons (MT) uranium as SNM, 215,000 MTU as source material, 10 curies (Ci) of cesium-137 (byproduct material) and various smaller quantities of other SNM and source and byproduct material.

The applicant is proposing to enrich the uranium-235 (^{235}U) isotope concentration in uranium up to 10 weight percent (wt%). The ^{235}U concentration in natural uranium is 0.71 wt%. However, the applicant is not expected to generate enriched product above 5 wt% ^{235}U as light water nuclear power reactors in the United States and power reactors abroad do not use fuel above this enrichment level nor are there any concrete plans to do so in the near future. In addition, the 2.5 ton UF_6 cylinders that the applicant is proposing to use to transport its product to its customers are not approved at the 10 wt% enrichment level. Nevertheless, the applicant's intent is to design the ACP to safely handle uranium enriched to 10 wt% in ^{235}U so that it can readily respond to any future demand for uranium in the 5 wt% and 10 wt% enrichment range. On this basis, NRC staff reviewed the application to ensure that the facility design could be safely operated with enrichments up to 10 wt% in ^{235}U. To allow NRC to confirm that no adverse safety or regulatory implications would result during transportation if the applicant began shipping uranium product enriched between 5 and 10 wt% in ^{235}U, the staff is imposing the following license condition:

> *USEC Inc. shall provide a minimum 60-day notice to NRC prior to initial customer product withdrawal of licensed material exceeding 5 wt. percent ^{235}U enrichment. This notice shall identify the necessary equipment and operational changes to support customer product shipment for these assays.*

The applicant has included technetium-99 (Tc-99) and transuranics in Table 1.2-1 of the LA (USEC, 2006c). These radionuclides may exist at the ACP in the form of sealed and unsealed sources, process contaminants and waste or material held in UF_6 cylinders from previous operations, or be introduced into the ACP as a result of processing former Soviet Union or recycled uranium. As indicated in Footnote F of Table 1.2-1 of the LA (USEC, 2006c), the applicant has committed to comply with the requirements of American Society of Testing and Materials (ASTM) ASTM C787-03, "Standard Specification for Uranium Hexafluoride for Enrichment" (ASTM, 2003), or ASTM C996-96, "Standard Specification for Uranium Hexafluoride for Enriched to less than 5% ^{235}U" (ASTM, 2004). These standards contain purity requirements of the uranium enrichment feed and enriched product. NRC staff considers the applicant's commitment to ASTM Standard C787-03 (ASTM, 2003) an acceptable means for ensuring that the Tc-99 and transuranic possession limits contained in Table 1.2-1 of the LA (USEC, 2006c) are not exceeded.

As stated above, the applicant identified the elemental name, maximum quantity, and specifications, including the chemical and physical forms, of the licensed material that the applicant proposes to acquire, deliver, receive, possess, produce, use, transfer, or store. The applicant also identified the isotopic content and amount of enrichment by weight percent of the licensed material. The information provided by the applicant meets the guidance in Section 1.2.4.3(3) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.2.3.5 Authorized Uses

The application is for the issuance of licenses under 10 CFR Parts 30, 40, and 70. Table 1.2-2 of the LA (USEC, 2006c) lists the authorized uses of the SNM and source and byproduct material.

The applicant is proposing to use SNM and source material in the enrichment of uranium. The uranium enrichment services would be sold to clients for the production of low-enriched uranium would be sold to customers to be ultimately used in the manufacture of fuel for commercial nuclear power plants. Byproduct material would be used in various applications such as in instrument calibration sources.

The applicant proposed a 30-year license term. The applicant also requested approval of a classified-matter facility clearance, under 10 CFR Part 95, by submitting a "Security Plan for the Protection of Classified Matter" (USEC, 2006d).

As stated above, the applicant provided a summary, non-technical narrative description for each activity or process in which the applicant proposed to acquire, deliver, receive, possess, produce, use, process, transfer, or store licensed material. The authorized uses of licensed material proposed for the facility are described and are consistent with the Atomic Energy Act of 1954, et seq. The description is also consistent with more detailed process descriptions submitted as part of the ISA Summary (USEC, 2006b), as reviewed under Chapter 3 of this SER. The information provided by the applicant meets the guidance in Section 1.2.4.3.(4) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.2.3.6 Special Exemptions or Special Authorizations

In Section 1.2.5 of the LA (USEC, 2006c), the applicant has requested seven exemptions to regulations and a special authorization.

The following two exemption requests and one request for an alternative method to the applicable 10 CFR Part 20 requirements are identified in Sections 1.2.5 and 4.8 of the LA (USEC, 2006c):

- Under the regulations in 10 CFR 20.1904, the licensee must ensure that each container of licensed material bears a durable, clearly visible label identifying it as containing radioactive material. The applicant has requested that UF_6 feed, product, and depleted uranium cylinders, which will be routinely transported inside the PORTS reservation boundary between ACP locations and/or storage areas at the ACP, be exempt from radioactive material container labeling requirements of 10 CFR 20.1904 as these will be readily identifiable due to their size and unique construction. Qualified radiological workers will attend UF_6 cylinders during movement. NRC staff grants the requested exemption based on the evaluation in Chapter 4 of this SER.

- The applicant has requested that containers located in Restricted Areas within the ACP be exempt from container labeling requirements of 10 CFR 20.1904. In such areas, one sign stating that every container may contain radioactive material will be posted. By procedure, when containers are to be removed from contaminated or potentially contaminated areas, a survey will be performed to ensure that contamination is not spread around the reservation. NRC staff grants the requested exemption based on the evaluation in Chapter 4 of this SER.

- Under the regulations in 10 CFR 20.1601(a), a licensee must control access to high radiation areas by using control devices to reduce exposures or energize alarms, or by locking entryways. In addition, the regulations allow an applicant to request approval for alternative methods for controlling access to high radiation areas. In lieu of the personnel access control requirements of 10 CFR 20.1601(a), the applicant has proposed for each High Radiation Area with a radiation reading greater than 0.1 roentgen equivalent man per hour (rem/hour) at 30-centimeters (cm) but less than 1 rem/hour at 30 cm to post "Caution, High Radiation Area" and to control entrance into the area by a Radiation Work Permit. The applicant proposes to maintain physical and administrative controls to prevent inadvertent or unauthorized access to High and Very High Radiation Areas. NRC staff approves the requested alternative method based on the evaluation in Chapter 4 of this SER.

The following request for exemption from the applicable 10 CFR 70.50 reporting requirement is identified in Sections 1.2.5 and 11.6.3 of the LA (USEC, 2006c):

- The 10 CFR 70.50(c)(2) reporting criteria require that the ACP submit a written follow-up report within 30 days of the initial report required by 10 CFR 70.50(a) or (b) or by 10 CFR 70.74 and Appendix A of Part 70. In lieu of the 30-day requirement described in 10 CFR 70.50(c)(2), the applicant has proposed to submit the required written reports within 60 days of the initial notifications. According to the applicant, an additional 30 days would provide more time to complete the required root cause analyses and would result in fewer supplemental reports thereby reducing regulatory burden and confusion.

  Because of the comprehensive nature of event follow-up reports, NRC staff would expect a 30-day event report to often be incomplete because event analysis and root cause determinations may often not be completed within 30 days. In these cases, a supplemental report would need to be submitted when information is complete. In recognition of this, for nuclear power reactors, NRC had revised 10 CFR Part 50 to allow 60 days for submitting event follow-up reports (*Federal Register*, October 25, 2000, Volume 65, No. 207, pp. 63769-63789). Considerations mentioned in connection with revising Part 50 included that the increased time would allow for completion of required engineering evaluations after event discovery, provide for more complete and accurate event reports, and result in fewer event report revisions and supplemental reports. Similar reasoning was provided when NRC staff granted exemptions to the Paducah and Portsmouth GDPs to allow 60 days for submitting event follow-up reports in lieu of the 30-day requirement in 10 CFR 76.120(d)(2) (*Federal Register*, November 12, 2002, Volume 67, No. 218, pp. 68699-68701). 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 granting this exemption will not endanger life or property or the common defense and security. Therefore, NRC staff grants the requested exemption.

The following exemption from the applicable 10 CFR 40.31(l) and 10 CFR 70.22(n) liability insurance requirements is identified in Section 1.2.5 of the LA (USEC, 2006c):

- The regulations in 10 CFR 40.31(l) and 70.22(n) require an applicant for a license to use SNM in a uranium enrichment facility to include the applicant's provisions for liability insurance. Specific liability insurance requirements for a uranium enrichment facility are provided in 10 CFR 140.13b. In the event that private liability insurance is unavailable, the applicant is proposing to use the existing indemnity agreement with DOE contained in its site lease pursuant to DOE's authority in Section 170d of the Atomic Energy Act. The applicant provided a letter from DOE, dated March 7, 2005 (DOE, 2005), that provided draft language from the lease agreement in which DOE will indemnify the ACP. Under the terms of the draft lease agreement, DOE will provide indemnity for the ACP unless liability insurance is commercially available at commercially reasonable rates.

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 DOE's indemnification of the ACP is an adequate alternative means for meeting the intent of the requirements of 10 CFR 140.13b. NRC staff's granting of the requested exemption is contingent upon the applicant providing a copy of the signed lease with the appropriate indemnification language included and documentation of the DOE determination that liability insurance is unavailable from commercial sources at commercially reasonable rates. If DOE does not provide acceptable indemnification, the applicant will be required to obtain full liability insurance. A license condition will be included in the license that will confirm DOE's indemnification of the ACP or proof of liability insurance. This license condition is discussed further in Section 1.2.3.3.3 of this SER.

The following exemption from the applicable 10 CFR 40.36 and 10 CFR 70.25 decommissioning funding requirements is identified in Sections 1.2.5 and 10.10.4 of the LA (USEC, 2006c):

- Based on its proposed phased approach in building and installing centrifuges and generating depleted uranium tails, the applicant requested an exemption to 10 CFR 40.36 and 10 CFR 70.25 to allow incremental funding for decommissioning based on the expected number of centrifuges to be built and installed and on the expected amount of depleted uranium tails to be generated annually in a forward-looking manner. As discussed in Section 10.10.4 of the LA (USEC, 2006c), the applicant stated that it would initially provide funding for the projected cost of facility decontamination and decommissioning, assuming operation at full capacity, except for the following:

    - Decontamination and removal of the centrifuges are incrementally funded on an annual forward-looking basis; and

    - The UF_6 tails are funded as they are generated on an annually forward-looking basis.

    NRC staff will review the initial cost estimate and the expected financial instrument before the applicant takes possession of licensed material. NRC staff will also review all subsequent revisions to the cost estimate and financial instruments. A more detailed discussion of this exemption request is contained in Chapter 10 of this SER.

    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, NRC staff grants the requested exemption. 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 Chapter 10 of this SER.

The following exemption from the requirements of 10 CFR 70.24 addressing criticality monitoring is identified in Section 1.2.5 of the LA (USEC, 2006c) and Section 3.10.6 of the ISA Summary (USEC, 2006b) and discussed in Section 5.4.4 of this LA (USEC, 2006c). The applicant has specifically requested an exemption from criticality monitoring of the UF_6 cylinder storage yards.

- 10 CFR 70.24, Criticality Accident Requirements, requires that licensees authorized to possess SNM in a quantity exceeding 700 grams of contained ^{235}U shall maintain in each area in which such licensed SNM is handled, used, or stored, a monitoring system capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within 1 minute. The applicant has demonstrated in Section 1.2.5 of the LA (USEC, 2006c) that based on several natural and instituted factors, the likelihood of a criticality event in the UF_6 cylinder storage yards is sufficiently low to justify this exemption request. NRC staff grants the requested exemption based on the evaluation in Chapter 5 of this SER.

In Sections 1.2.5 and 4.8.2.4 of the LA (USEC, 2006c), the applicant has requested the following Special Authorization:

- Surface Contamination Release Levels for Unrestricted Use – Items may be released for unrestricted use if the surface contamination is less than the levels listed in Table 4.6-1 of the LA (USEC, 2006c). NRC staff finds this Special Authorization acceptable. The bases for this determination are contained in Chapter 4 of this SER.

1.2.3.7 Security of Classified Matter

The purpose of this review is to verify that the applicant provided sufficient information to conclude that there is an adequate classified matter plan (CMP) for the protection of classified matter at the proposed facility to be located in Piketon, Ohio, and a facility clearance can be issued.

1.2.3.7.1 Regulatory Requirements

10 CFR 70.22(m) requires applicants to contain a full description of an applicant's security program to protect against theft, and to protect against unauthorized viewing of classified enrichment equipment, and unauthorized disclosure of classified matter in accordance with the requirements of 10 CFR Part 95.

1.2.3.7.2 Regulatory Guidance and Acceptance Criteria

The applicant's CMP (USEC, 2006d) was reviewed for compliance with the requirements of 10 CFR Part 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" (NRC, 1999).

1.2.3.7.3 Staff Review and Analysis

The applicant submitted its CMP entitled "Security Plan for the Protection of Classified Matter" (USEC, 2006d) as Chapter 2 of its "Security Program for the American Centrifuge Plant" (USEC, 2006d). Chapters 1 and 3 of the Security Program for the ACP (USEC, 2006d) contain the applicant's "Physical Security Plan for the Protection of Special Nuclear Material of Low Strategic Significance" and "Physical Security Plan for the Transportation of Special Nuclear Material of Low Strategic Significance," respectively. NRC staff's evaluations of the applicant's security plans for SNM and transportation are contained in Appendix I and Appendix J of this SER, respectively. The applicant's CMP (USEC, 2006d) outlines 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 Part 95.

The ACP will be primarily located in the former Gas Centrifuge Enrichment Plant facilities and in several new buildings on the southwest portion of the Portsmouth reservation. The ACP facilities will be located within the DOE reservation boundary and it will maintain its own Controlled Access Area boundary. Access controls for the ACP will be in addition to those provided to limit access to the DOE reservation and DOE boundary post. The West Gate access/portal will be the primary access point for the ACP.

NRC has reviewed the applicant's CMP for the ACP and found it to satisfy the requirements of 10 CFR Part 95. The applicant has made commitments that meet the requirements of 10 CFR Part 95 by providing an acceptable CMP that establishes controls to ensure that classified matter is used, processed, stored, reproduced, transmitted, transported, and destroyed only under conditions that will provide adequate protection and prevent access by unauthorized persons. By meeting these requirements, the applicant complies with the requirements of 10 CFR 70.22(m).

At the time NRC reviewed the applicant's CMP (USEC, 2006d), a specific facility for use and storage of classified matter had not been identified other than as provided under DOE authority. Because of this, the staff will impose the following license condition:

> *USEC Inc. shall not use, process, store, reproduce, transmit, handle, or allow access to classified matter except provided by applicable personnel and facility clearances as required under 10 CFR Part 95.*

It is noted that this license condition will not apply to any DOE-authorized activities involving

classified matter in existing buildings and areas that fall within the boundaries of the ACP.

The information provided by the applicant's CMP meets the requirements of 10 CFR Part 95 and the guidance in "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" (NRC, 1999) and Section 1.2.4.3.(6) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable. However, NRC's authorization for the applicant to begin implementation of its CMP is contingent upon an NRC inspection and finding prior to receipt of classified matter that its classified matter program at the ACP is being implemented in accordance with its CMP (USEC, 2006d). The licensee shall not use, process, store, reproduce, transmit, handle, or allow access to classified matter except provided by applicable personnel and facility clearances as required under 10 CFR Part 95.

1.2.4 EVALUATION FINDINGS

The staff reviewed the institutional information for the proposed ACP, according to Section 1.2 of the Standard Review Plan. The applicant has adequately described and documented the corporate identity, structure, and financial information, and is in compliance with those parts of 10 CFR 30.32, 10 CFR 40.31, 10 CFR 70.22, and 10 CFR 70.65 related to institutional information.

The staff reviewed information related to the applicant's financial qualifications. To ensure the applicant meets the financial qualifications requirements for construction and operation of the facility, the staff will impose the following license conditions:

> *Construction of each incremental phase of the ACP shall not commence before funding for that increment is available or committed. Of this funding, USEC Inc. must have in place before constructing such increment, commitments for one or more of the following: equity contributions from USEC Inc., affiliates and/or partners, along with lending and/or lease arrangements that solely or cumulatively are sufficient to ensure funding for the particular increment's construction costs. USEC Inc. shall make available for NRC inspection documentation of both the budgeted costs for such phase and the source of funds available or committed to pay those costs.*
>
> *Operation of the ACP shall not commence until USEC Inc. has in place either: (1) long term contracts lasting five years or more that provide sufficient funding for the estimated cost of operating the facility for the five year period; (2) documentation of the availability of one or more alternative sources of funds that provide sufficient funding for the estimated cost of operating the facility for five years; or (3) some combination of (1) and (2).*

The staff reviewed the information provided by the applicant on liability insurance. This information meets the requirements of 10 CFR 140.13b. Because full liability insurance coverage will not be provided until prior to receipt of licensed material, NRC staff is imposing the license condition provided in Section 1.2.3.3.3 of this SER:

> *USEC Inc. shall provide to the Commission, at least 120 days prior to the planned date for obtaining licensed material, documentation of any liability*

insurance required to be obtained by USEC Inc. under its lease with DOE for the ACP by that time or, alternatively, the status of USEC Inc.'s efforts to obtain any such liability insurance. During the time that USEC Inc. is engaged in efforts to obtain liability insurance, USEC Inc. shall provide the Commission with status reports regarding those efforts. The status reports shall be submitted at a frequency of at least once every six months following issuance of a license. USEC Inc. shall notify the Commission within 30 days upon receiving notification of denial or approval of commercial liability insurance for the ACP. If commercial liability insurance is required to be obtained under its lease with DOE, within 60 days of receiving notification of approval of commercial liability insurance, USEC Inc. shall provide proof of liability insurance coverage and a justification, for Commission review and approval, if USEC Inc. is proposing to provide less than $300 million of liability insurance coverage.

The staff reviewed the applicant's request for possessing uranium enriched in ^{235}U up to 10 wt%. If a license is issued and operations begin at the ACP, the applicant is not anticipated to initially produce uranium above 5 wt% in ^{235}U during the next several years since no demand for uranium above 5 wt% ^{235}U is anticipated. However, if the need for the ACP to generate uranium at enrichments between 5 and 10 wt% ^{235}U is created, then to allow NRC to confirm that no adverse safety or regulatory implications would result inside or outside the ACP (such as during transportation), the staff is imposing the following license condition:

USEC Inc. shall provide a minimum 60-day notice to NRC prior to initial customer product withdrawal of licensed material exceeding 5 wt. percent ^{235}U enrichment. This notice shall identify the necessary equipment and operational changes to support customer product shipment for these assays.

The staff reviewed the applicant's "Security Plan for the Protection of Classified Matter" (USEC, 2006d) and found it to satisfy the requirements of 10 CFR Part 95. Because a specific facility for use and storage of classified matter had not been identified other than as provided under DOE authority, staff is imposing the following license condition:

USEC Inc. shall not use, process, store, reproduce, transmit, handle, or allow access to classified matter except provided by applicable personnel and facility clearances as required under 10 CFR Part 95.

In addition, in accordance with 10 CFR 30.32, 10 CFR 40.31, and 10 CFR 70.22(a)(2) and (4), the applicant has adequately described the types, forms, and quantities and proposed purpose and authorized uses of licensed materials to be permitted at the facility. The applicant provided information on six exemption requests and two special authorizations related to radiation protection, criticality monitoring alarms, event reporting, liability insurance, and decommissioning funding that meet the requirements of 10 CFR 40.14 and 10 CFR 70.17. The applicant has also adequately described information related to FOCI, 10 CFR 40.38, and 10 CFR 70.40, and its plans to secure classified matter for a facility clearance under 10 CFR Part 95. The staff concludes that the applicant has met the requirements and acceptance criteria in Section 1.2.4.3 of NUREG-1520 (NRC, 2002).

1.3 SITE DESCRIPTION

The purpose of NRC's review of the applicant's site description is to evaluate whether the application adequately describes the geographic, demographic, meteorological, hydrologic, geologic, 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 ER, emergency plan (EP), and ISA Summary.

1.3.1 REGULATORY REQUIREMENTS

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).

1.3.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The acceptance criteria applicable to NRC's review of the site description section of the LA (USEC, 2006c) are contained in 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, 10 CFR 70.65(b)(1), and Section 1.3.4.3 of NUREG-1520 (NRC, 2002). Chapter 1 of NUREG-1520 is applicable to the ACP facility in its entirety.

1.3.3 STAFF REVIEW AND ANALYSIS

1.3.3.1 Site Geography

Section 1.3 of the LA (USEC, 2006c) describes the ACP's location and description, nearby roadways and bodies of water, and significant geographical features. Section 1.3 states that the ACP will be located on DOE-owned land in rural Pike County, a sparsely populated area in south-central Ohio. Specifically, the ACP will be located on the PORTS reservation in the former Gas Centrifuge Enrichment Program facilities. The PORTS reservation is in Pike County on the east side of the Scioto River approximately equidistant between Portsmouth and Chillicothe, Ohio. The Scioto River Valley is 1 mile west of the reservation. The Scioto River, approximately 2 miles west of the reservation, is a tributary of the Ohio River, and their confluence is approximately 25 miles south of the reservation. With the exception of the Scioto River flood plain, which is farmed extensively, the area around the reservation consists of marginal farmland and forested hills. The only other body of water located near the reservation is Lake White, which is located approximately 6 miles north of the reservation. With the exception of the Scioto River flood plain, which is farmed extensively, the area around the site consists of marginal farmland and forested hills.

Located adjacent to the site are two major four lane highways: U.S. Route 23, traversing north-south, and U.S. Route 32/124, traversing east-west, service the reservation. Commercial air transportation is provided through the Greater Cincinnati International Airport (approximately 100 miles west), the Port Columbus International Airport (approximately 75 miles north), or the Tri-State Airport (approximately 55 miles south-east). The Greater Portsmouth Regional Airport, serving private and charter aircrafts, is located approximately 15 miles southeast near Minford, Ohio, and the Pike County Airport, located just north of Waverly, is a small facility for private planes.

The entire PORTS reservation is marked and bounded by signs and fences (barbed wire in the wooded areas). Where roads cross the boundary, gates are in place to serve as barriers if

needed. PORTS reservation boundaries are identified in Figure 1.1-1 of the LA (USEC, 2006c). The reservation boundary will be the controlled area boundary specified in 10 CFR 70.61(f). Most buildings and activities at the site (including the ACP facilities) are located within the next level of control, a Controlled Access Area (CAA), surrounded by a security fence. Access to this fenced area will be gained only with approved identification. In addition, the ACP will be located within its own CAA. A topographic map of the PORTS reservation is provided in Figure 1.3-1 of the LA (USEC, 2006c).

1.3.3.1.1 Portsmouth GDP

The Portsmouth GDP, constructed in the 1950s, is located on the PORTS reservation in Piketon. NRC assumed regulatory oversight of the Portsmouth GDP from DOE in March 1997 under a 10 CFR Part 76 certificate issued to the United States Enrichment Corporation, a subsidiary of the applicant. Although enrichment operations at the PORTS ceased in May 2001, the United States Enrichment Corporation continues to maintain its NRC certificate to conduct certain remedial and recovery operations and to maintain the plant in standby mode.

1.3.3.1.2 Depleted Uranium Hexafluoride Conversion Facility

DOE plans to construct a depleted uranium hexafluoride (DUF_6) conversion facility on the PORTS reservation. It will be located north of the X-7725 facility. At the conversion facility, DOE will convert its inventory of DUF_6 to a stable oxide form of uranium. The Final Environmental Impact Statement (EIS) for this facility was issued and made available to the public in June 2004 (DOE, 2004). Construction of the facility began after the EIS was published.

1.3.3.1.3 Site Geography Conclusions

The applicant provided a summary describing the site geography, including its location relative to prominent natural and man-made features (such as mountains, rivers, airports, population centers, schools, and commercial and manufacturing facilities). The summary also described the site boundary and controlled area boundary. NRC staff reviewed information provided in the LA (USEC, 2006c) and ISA Summary (USEC, 2006b) on site geography, and finds the data used to be accurate. The applicant's descriptions are consistent with the more detailed information in the ISA Summary (USEC, 2006b), the ER (USEC, 2003a), and the EP (USEC, 2006a). The information the applicant provided is consistent with the guidance in Sections 1.3.4.3(1) and 1.3.4.3(5) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.3.3.2 Demographics

The nearest residential center and the closest town to the site is Piketon, located in Pike County, about 4 miles north of the site on U.S. Route 23, with a population of 1,907 in 2000. The largest town in Pike County is Waverly, about 8 miles north of the site, with a population of 4,433 in 2000. The total population within the 5 mile radius of the site was 5,836 in 2000.

The two school systems in the area are the Pike County Schools and the Scioto County Schools. However, only Pike County has school facilities within 5 miles of the facility: a private school, 2 elementary schools, one that also has a preschool included; one high school; and a vocational school. The combined enrollment of these schools for the year 2003-2004 year was

approximately 2,437. The total school population within five miles, including faculty and staff, was 2,718 for the 2003-2004 year. The proximity of these schools to the site and their enrollments are shown in Figure 1.3-3 of the LA (USEC, 2006c).

Four facilities within five miles of the PORTS reservation provide day care or schooling for preschool-aged children and after-school care for school-aged children. One facility had 114 registered children for school year 2003-2004 and is located in Piketon; the other, licensed to accommodate 70 children, is located near the PORTS reservation boundary. The third facility is consolidated in the numbers provided above. The locations of these facilities are shown in Figure 1.3-3 of the LA (USEC, 2006c).

Pike Community Hospital is the hospital closest to the site, located approximately 7.5 miles north of the facility on State Route 104 south of Waverly. The facility has 70 licensed beds. No other acute care facilities are located in Pike County. The location of Pike Community Hospital is shown in Figure 1.3-3 of the LA (USEC, 2006c). Adena Regional Medical Center and Pike Community Hospital operate as urgent care facilities, both are located approximately 7.5 miles north of the site.

Five licensed nursing homes are located near the site. Those are located in or near Piketon, one in Beaver, and one in Wakefield; four are located within five miles of the site. The largest of these facilities is a 193-bed facility in Piketon.

No significant recreational areas are on the site; recreational activities for employees are held off site.

Off-site recreational areas include the Brush Creek State Forest, a 0.5 square mile portion of which is within five miles southwest of the PORTS reservation. Usage of this area is extremely light and is estimated to be 20 persons/year, primarily hunters and mushroom pickers. The location of Brush Creek State Forest is identified in Figure 1.3-3 of the LA (USEC, 2006c). Lake White State Park, identified in Figure 1.3-3 of the LA (USEC, 2006c), is located approximately six miles north of the site. It offers recreation such as boating, fishing, water skiing, and swimming. Usage is occasionally heavy and concentrated on the 92 acres of land closest to the lake.

Land within five miles of the DOE reservation is used primarily for farms, forest, and rural residences. About 25,430 acres of farmland, including cropland, wooded lot, and pasture, lie within five miles of the reservation. The cropland is located mostly on or adjacent to the Scioto River flood plain and is farmed extensively, particularly with grain crops. The hillsides and terraces are used for cattle pasture. Both beef and dairy cattle are raised in the area.

The only significant industry in the vicinity is located in an industrial park south of Waverly. The industries include a cabinet manufacturer and an automotive parts manufacturer. These industries do not present any potential hazards to ACP operations.

Approximately 24,000 acres of forest lie within five miles of the reservation. This includes some commercial woodlands and a very small portion of Brush Creek State Forest.

No known public or private water is withdrawn from the Scioto River downstream of the ACP.

The applicant provided a summary of demographic information based on the most recent census data that showed the population distribution as a function of distance from the proposed facility. The applicant's descriptions are consistent with the more detailed information in the ISA Summary (USEC, 2006b), the ER (USEC, 2003a), and the EP (USEC, 2006a). The information the applicant provided is consistent with the guidance in Sections 1.3.4.3(2) and 1.3.4.3(5) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.3.3.3 Meteorology

Section 1.3.3 of the LA (USEC, 2006c) and Section 1.3 of the ISA Summary (USEC, 2006b) provide a meteorological description of the site and its surrounding area. Section 1.3.3.1 of the LA (USEC, 2006c) and Section 1.3.1 of the ISA Summary (USEC, 2006b) state that July is the hottest month, with an average monthly temperature of 23.3 °C (74 °F), and January is the coldest month with an average temperature of −1.1 °C (30 °F). The highest and lowest daily temperatures from 1951 to 2002 were 39.4 °C (103 °F) and −35 °C (−31 °F) on July 14, 1954, and January 19, 1994, respectively. The average annual precipitation at Waverly, Ohio, for the period from 1951 to 2002 was 102 cm (40 in). The average annual snowfall for the area is 53.6 cm (21.1 in), based on the 1951–2002 data. During that time period, the maximum monthly snowfall was 64.5 cm (25.4 in), occurring in January 1978. The prevailing winds at the site blow from the south-southwest and southwest directions to the north-northeast and northeast directions. On the average, from 1950 to 2002, 18 tornadoes per year were reported in Ohio, but the total varies widely from year to year (e.g., 63 in 1992 and 0 in 1988). Pike County has had three tornadoes since 1950.

1.3.3.3.1 High Winds and Hurricanes

Information about high winds at the proposed ACP is provided in Section 2.5.1.3 of the ISA Summary (USEC, 2006b). The estimated high wind for a 250-year return period, based on a site-specific study, is 126 km/h [78 mph], and 161 km/h [100 mph] for a 20,000-year return period. These estimated values are acceptable to the staff because these values are consistent with the high wind and tornado hazard curves proposed by Coats and Murray (Coats, 1985) for the PORTS reservation.

Because the proposed ACP is not located near a coastal area, hurricanes affecting the coastal area will have no effect on the performance of the plant. Consequently, consideration of hurricane wind hazards on the design of the proposed ACP is not needed. Heavy rain is considered in Section 1.3.3.3.4, "Extreme Precipitation," of this SER.

Based on review of the information concerning high winds, the staff concludes that the high-wind hazards and the associated design-basis straight-line winds have been addressed acceptably because the data used for the assessments were from a recognized source and the method used for analyzing high-wind hazard is a commonly used and accepted method.

1.3.3.3.2 Tornado Hazard and Tornado-Generated Missiles

Information about the tornadoes at the PORTS reservation area is provided in Section 1.3.3.3 of the LA (USEC, 2006c) and Sections 1.3.3 and 2.5.1.3 of the ISA Summary (USEC, 2006b).

According to a National Oceanic and Atmospheric Administration (NOAA) (NOAA, 2006) database, the State of Ohio had an average of 18 reported tornadoes per year. Three tornadoes since 1950 are known to have occurred in Pike County (NOAA, 2006). The applicant addressed the potential tornado hazards in Sections 2.5.1.3 and 6.1.1.7.3, and Accident Sequence SR7-3 of the ISA Summary (USEC, 2006b). The applicant used a formula proposed by Fujita (Coats, 1985) to determine the annual probability of tornado occurrence at the site using the tornado data of Pike County and the surrounding five counties. The results of this calculation indicated that the annual probability of a tornado striking certain buildings within the PORTS reservation meets the performance requirements of 10 CFR 70.64(a)(2) (i.e., highly unlikely).

Based on the analysis result, the applicant concluded that the calculated tornado probability agrees with the severe winds and tornado hazard curves recommended for the Portsmouth, Ohio, area by Coats and Murray (Coats, 1985). The tornado hazard curve proposed by Coats and Murray (Coats, 1985) suggests the estimated tornado wind speeds of 193 km/h [120 mph] and 241 km/h (150 mph) for 30,000- and 100,000-year return periods.

The staff verified that the frequency of a tornado hitting any facility building (except the cylinder storage yards) was less than 10^{-5} /year or "highly unlikely" (Spivack, 2006). Staff reviewed the information presented in the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b) and finds the tornado hazard analysis conducted by the applicant is acceptable and the tornado hazards at the site have been determined appropriately.

1.3.3.3.3 Temperature Extremes

Information about the temperature at the PORTS reservation area, where the proposed ACP is located, is provided in Sections 1.3.3.1 and 1.3.3.3 of the LA (USEC, 2006c). The same information is also presented in Sections 1.3.1 and 1.3.3 of the ISA Summary (USEC, 2006b).

The applicant provided the monthly average and extreme temperatures published by the NOAA from the weather stations in Waverly and Piketon, Ohio, for various measurement periods. The observed temperature extremes for Piketon, Ohio, from 1951 to 2002 range from −35 to 39.4 °C (−31 to 103 °F).

There is a meteorological tower at the PORTS reservation area. This tower has been in use since 1995. The temperature extremes measured at the site are determined to be within the range of the temperature extremes measured at the Waverly and Piketon, Ohio, weather stations.

Staff reviewed the temperature information and finds the information acceptable because recognized data sources were used and the temperature extremes are properly determined.

1.3.3.3.4 Extreme Precipitation

Sections 1.3.3.1 and 1.3.4.3.1 of the LA (USEC, 2006c) and Sections 1.3 and 1.4.3.1 of the ISA Summary (USEC, 2006b) discussed the precipitation at the proposed ACP site. Between 1951 and 2002, the average annual precipitation was 102 cm (40 in), and the greatest daily

precipitation was 12.4 cm (4.9 in).

Table 1.3-2 of the LA (USEC, 2006c) listed the precipitation as a function of recurrence interval for various durations. These data were from the National Weather Service for durations from 30 minutes to 24 hours, except for the precipitations for the 10,000-year return period. The precipitations for the 10,000-year return period were extrapolated based on the data from 1 to 100 years using a nonlinear least-squares method.

Staff reviewed the information presented in the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b) concerning extreme precipitation and finds the information acceptable because recognized data sources, such as the National Weather Service, were used. The least-squares method used to estimate precipitations for a return period of 10,000 years is acceptable to the staff because it is a recognized statistical method for analyzing the type of data presented in Table 1.3-2 of the LA (USEC, 2006c).

1.3.3.3.5 Snow

Sections 1.3.3.1 and 1.3.4.3.1 of the LA (USEC, 2006c) briefly discussed the regional and local snowfall. The same information is also presented in Section 1.4.3.1 of the ISA Summary (USEC, 2006b). Between 1951 and 2002, the average annual and the maximum monthly snowfalls recorded for the area were 53.6 cm (21.1 in) and 64.5 cm (25.4 in). The recorded maximum monthly snowfalls for the surrounding three cities are 87.4 cm (34.4 in) for Columbus, Ohio; 100.5 cm [39.5 in] for Charleston, West Virginia; and 72.1 cm (28.4 in) for Louisville, Kentucky. The LA (USEC, 2006c) indicates the Charleston, West Virginia, maximum monthly snowfall, equivalent to 10.1 cm (3.95 in) of rainfall, is used for the PORTS reservation area.

According to Figure 7-1 in Section 7.2 of Structural Engineering Institute (SEI)/American Society of Civil Engineers (ASCE) 7-02 (SEI/ASCE, 2003), the ground snow load at a 2-percent annual probability of being exceeded (i.e., a 50-year mean recurrence interval) at the proposed ACP site is approximately 0.96 kPa [20 psf]. The value the applicant used for the reservation area is consistent with the value suggested by the SEI/ASCE 7-02 standard and, therefore, is acceptable to the staff. For the reference to this standard, see Sections 1.3.3.1 and 1.3.3.3 of the LA (USEC, 2006c). This value was used as the design basis ground snow load for the Process Buildings X–3001 and X–3002, as stated in the LA (USEC, 2006c). NRC staff verified the use of the design basis ground snow load for the design of these two Process Buildings during an on-site review conducted as part of NRC's application review of the applicant's American Centrifuge Lead Cascade Facility LA (USEC, 2003c).

1.3.3.3.6 Lightning

Section 1.3.3.3 of the LA (USEC, 2006c) and Section 1.3.3 of the ISA Summary (USEC, 2006b) described the potential of lightning strikes at the proposed ACP site. The applicant indicated that between 1989 and 1998, the proposed ACP site has had an average of 36 thunderstorms per year that may produce lightning strikes. The applicant points out further in the LA (USEC, 2006c) and ISA Summary (USEC, 2006b) that the proposed ACP and the associated power systems are designed and built with heavy grounding or lightning protection to handle lightning strikes.

The design approach used by the applicant to protect the proposed ACP from lightning effects

is acceptable because, as stated in ISA Summary (USEC, 2006b), the buildings for the proposed ACP are grounded and are adequately protected from lightning strikes.

1.3.3.3.7 Meteorology Conclusions

The applicant provided appropriate meteorological data, including a summary of design-basis values for accident analysis of maximum snow loads, and probable maximum precipitation, as presented in the ISA Summary (USEC, 2006b). The applicant also provided appropriate design-basis information for lightning, high winds, tornadoes, hurricanes, extreme precipitation, and temperature extremes. The applicant's descriptions are consistent with the more detailed information in the ISA Summary (USEC, 2006b), the ER (USEC, 2003a), and the EP (USEC, 2006a). The information the applicant provided is consistent with the guidance in Sections 1.3.4.3(3) and 1.3.4.3(5) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.3.3.4 Hydrology

Section 1.3.4 of the LA (USEC, 2006c) describes the surface hydrology on and around the PORTS reservation. The PORTS reservation is located near the southern end of the Scioto River basin, which has a drainage area of 6,517 square miles. The headwaters of the Scioto River form in Auglaize County in north central Ohio. The river flows 235 miles through nine counties in Ohio, and through the cities of Columbus, Circleville, Chillicothe, and Portsmouth. At Portsmouth, in Scioto County, the river empties into the Ohio River at river mile 356.5. The slope of the Scioto River channel averages about 1.7 ft/mile between Columbus and Portsmouth. The river flows measured at Higby, Ohio, from 1930 to 1991 ranged from 177,000 cubic feet per second (cfs) on January 23, 1937, to 244 cfs on October 23, 1930, and average 4,654 cfs. The 1937 flood had a peak water elevation of 593.7 ft above mean sea level.

Water used at the site normally comes from groundwater. Currently, all water is supplied by wells in the Scioto River alluvium. These wells are located near the east bank of the Scioto River, downstream from Piketon. Four well fields have the capacity to reliably supply between 36.4 and 40.2 cfs.

The ACP's nominal elevation is 670 ft, which is about 113 ft above the normal stage of the Scioto River. Both groundwater and surface water at the site are drained from the plant site by a network of tributaries of the Scioto River. The top-of-slab floor elevations for the ACP facilities are at approximately 671 ft above minimum sea level. Storm water that falls at the site is drained to local Scioto River tributaries by storm sewers. The flow of storm water is further controlled by a series of holding ponds downstream from the storm sewer outfalls. The perimeter road, as shown in Figure 1.3-6 of the LA (USEC, 2006c), serves as a hydrologic boundary that prevents storm water runoff from backing up into the ACP facility.

To assess whether failures of the local dams could conceivably jeopardize the safety of the ACP, the applicant considered holding ponds, lagoons, and retention basins formed by these dams in the local drainage analysis. The surface elevations of all but the X-611B lagoon are well below the 670-ft minimum grade elevation of the ACP facilities. The water elevation of the X-611B sludge lagoon at 668.8 ft is close to the 670-ft minimum grade elevation at the ACP facility. The elevation of the top of the dam forming the lagoon is 676.3 ft and exceeds the 670-ft minimum. However, when the conservative estimate of flood wave height (4/9 of the dam height) is used, the flood elevation resulting from a break in the dam would be only 652.8 ft.

The nominal, top-of-grade elevation at the site is 670 ft, about 99 ft above the probable maximum flood (PMF) plus wind wave activity flood stage of 571 ft. The top-of-slab floor elevation for the ACP facility is at approximately 671 ft. The Scioto River during a PMF superimposed with wind wave activity; therefore, would not inundate these buildings.

The PORTS reservation water supply facility near the Scioto River, pump house X-608, and groundwater well fields, may expect flooding. Though the well fields are designated to operate during floods, the impacts of flooding on the ACP cooling system would not result in a release of UF_6 or a criticality. Closing strategic valves can isolate the enrichment process, and during severe conditions all or part of the cascade can be shut down. Therefore, flooding of the reservation water supply will not adversely affect plant safety.

The applicant provided a summary description of the site hydrology and cites the design-basis flood event for which the facility may be safely shut down. The applicant's hydrological data are sufficient to assess site flooding hazards and ground and surface water impacts. The applicant's descriptions are consistent with the more detailed information in the ISA Summary (USEC, 2006b), the ER (USEC, 2003a), and the EP (USEC, 2006a). The information the applicant provided is consistent with the guidance in Sections 1.3.4.3(4) and 1.3.4.3(5) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.3.3.5 Geology

1.3.3.5.1 Seismic Hazards

Seismic hazards are discussed in Section 1.3.6 of the LA (USEC, 2006c) and Section 1.6 of the ISA Summary (USEC, 2006b).

The following areas concerning the seismic hazard applicable to the safety analysis and design of the proposed facility were reviewed:

- Seismic source characterization;
- Ground motion attenuation;
- Seismic hazard calculation;
- Development of site-specific spectra; and
- Surface faulting.

1.3.3.5.1.1 Seismic Source Characterization

<u>Geological and Tectonic Settings</u>

Section 1.3.6 of the LA (USEC, 2006c) and Section 1.6 of the ISA Summary (USEC, 2006b) provide a description of the local and regional geological and tectonic settings. The LA (USEC, 2006c) notes that the PORTS reservation area is located within the Interior Low Plateaus physiographic province, bordered on the north and west by the Central Lowlands physiographic province and on the south and east by the Appalachian Plateau physiographic province. Bedrock beneath the PORTS reservation area consists of relatively flat-lying and unfaulted carbonate and clastic strata of Paleozoic age. The region also contains unconsolidated Quaternary lacustrian deposits related to Pleistocene glaciation of eastern North America and preglacial alluvial and fluvial deposits related to the ancient Portsmouth River.

The Interior Low Plateaus physiographic province lies within the stable craton of the North American tectonic plate and just to the west of the Appalachian orogenic belt. Active tectonic deformation of the Appalachians ended in the Permian or early Triassic (approximately 240 million years ago), as the orogen was rifted open in response to the breakup of Pangea and formation of the Atlantic Ocean basin. The low levels of earthquake activity in the region, including the Appalachians, are generally considered to be associated with preexisting zones of weakness in the crust that formed in the distant geologic past. These zones of weakness are characterized by deeply buried and poorly characterized faults, some of which accomplish a periodic release of strain that builds up continually in the North American continental plate. At the PORTS reservation area, postulated earthquakes that could impact safe operation of the proposed facility are associated with zones of crustal weakness in western Ohio, the Appalachians, and the New Madrid seismic zone.

Historical Seismicity

Section 1.3.6.5 of the LA (USEC, 2006c) and Section 1.6.5 of the ISA Summary (USEC, 2006b) provide a brief summary of historical seismicity at the site. The LA (USEC, 2006c) notes that no historical earthquakes have occurred within a 25-mi radius of the site. Within 50 mi of the site, the largest historical earthquake had an epicenter intensity of IV on the Modified Mercalli scale, which is roughly equivalent to a peak ground acceleration of approximately 0.02g at the site.

A summary of the state historic seismicity compiled by the Ohio Department of Natural Resources, Division of the Geological Survey, "Earthquakes and Seismic Risk in Ohio" (Ohio, 2000), indicates low levels of historic seismicity at the PORTS reservation area. It was reported also that more earthquakes have occurred in western Ohio than in other areas of the state. At least 40 felt earthquakes have occurred in western Ohio since approximately 1875, although most of these earthquakes caused little or no damage. The exceptions were two earthquakes in 1937, March 2 and March 9, which caused some damage in Anna, Ohio (e.g., toppled chimneys, cracked plaster, broken windows, and structural damage to buildings).

Northeastern Ohio has experienced approximately 20 felt earthquakes since 1836. Most of these events were small and caused little or no damage. One earthquake with a body wave magnitude of 5.1 struck on January 31, 1986, however. This earthquake occurred in northeastern Ohio and caused minor to moderate damage, including broken windows and cracked plaster, in the epicentral area located within Lake and Geauga Counties. Southeastern Ohio has been the site of less than 10 felt historic earthquakes. Earthquakes in 1901 near Portsmouth (Scioto County), in 1926 near Pomeroy (Meigs County), and in 1952 near Crooksville (Perry County), Ohio, caused minor to moderate damage. The LA (USEC, 2006c) notes that the peak ground motion recorded at the PORTS reservation area was 0.005g, recorded in 1955.

The probabilistic seismic hazard study of the site conducted by Risk Engineering, Inc., "Seismic Hazard Evaluation for the Portsmouth Gaseous Diffusion Plant" (Risk Engineering, 1992), provides the basis for the current seismic hazard at the ACP site. The Risk Engineering, Inc. study relies directly on seismic source characterizations provided by the Electric Power Research Institute (EPRI) (EPRI, 1988) and Lawrence Livermore National Laboratory (LLNL), "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains" (NRC, 1994), which are seismic hazard studies for the eastern United States. Earthquake source characteristics associated with the seismic zones and historic

seismicity discussed previously are consistent with information used in both the EPRI and LLNL studies. The seismic hazard spectra developed from the EPRI and LLNL studies form the basis for the development of the 1,000-year design basis earthquake (DBE) response spectrum for the proposed ACP. A more recent probabilistic seismic hazard study was conducted by ECS, LLP (ECS), "Submittal of Additional Information Regarding Seismic Analysis for the American Centrifuge Plant" (USEC, 2006e), for the applicant who subsequently forwarded the report to NRC by letter dated February 2, 2006. The ECS study relied on seismic source characterizations provided by the U.S. Geological Survey (USGS), which were documented in Frankel, et al., "National Seismic Hazard Maps: Documentation" (Frankel, 1996) and "Documentation for the 2002 Update of the National Seismic Hazard Maps" (Frankel, 2002). The results from the ECS study form the basis for the development of the 10,000-year DBE response spectrum for the proposed ACP.

NRC staff has previously accepted the LLNL and EPRI data, seismic sources, seismic hazard methods, and results in Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion" (NRC, 1997), for use at sites in the central and eastern United States. Thus, NRC staff concludes that applying the LLNL and EPRI hazard results is technically sound. Staff also concludes that applying the ECS hazard results, which were based on the USGS seismic source characterization, is technically sound. Therefore, information about seismic source characterization presented in the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b) is acceptable and it demonstrates compliance with the regulatory requirements in 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, and 10 CFR 70.65(b)(1).

1.3.3.5.1.2 Ground Motion Attenuation

Seismic hazards used to define the uniform hazard spectra at the PORTS reservation area are based on the LLNL and EPRI probabilistic seismic hazard studies. Ground motion attenuation functions used in the analyses and the relative weights given each model for computing the hazard are described in detail in Risk Engineering, Inc. (Risk Engineering, 1992). In addition, both the LLNL and EPRI probabilistic seismic hazard studies provide methods and results to develop site amplification factors that convert the hard rock uniform hazard spectra to soil hazard spectra. Soil profiles were chosen for till-like shallow soils that are typical for the Central Lowlands physiographic province. The probabilistic seismic hazard study conducted by ECS (USEC, 2006e) incorporates ground motion attenuation functions that are identical to those used by the USGS to develop the national seismic hazard maps (Frankel, et al., 2002) for the central and eastern United States.

Ground motion attenuation models used in the LLNL and EPRI studies provide representative and accurate models of ground motion attenuation characteristics in the central and eastern United States. Both studies have captured diverse opinions in the scientific community. NRC staff previously accepted the LLNL and EPRI ground motion modeling for sites in the central and eastern United States (NRC, 1997). In addition, ground motion attenuation models used in the ECS study (USEC, 2006e) provide representative and accurate models of ground motion attenuation characteristics in the central and eastern United States. Application of these models to the PORTS reservation area hazard assessment, and consequently, to the proposed ACP design is considered technically sound. As such, the information about ground motion attenuation presented in the LA (USEC, 2006c) is acceptable and demonstrates compliance with the regulatory requirements in 10 CFR 70.65(b)(1).

1.3.3.5.1.3 Seismic Hazard Calculation

The probabilistic studies were performed using the LLNL and EPRI seismic hazard methodologies (Risk Engineering, 1992). The LLNL and EPRI results for probabilistic seismic hazard were combined according to the methodologies described in the DOE–STD–1024–92, "Guidelines for the Use of Probabilistic Seismic Hazard Curves at Department of Energy Sites," (DOE, 1992) to obtain uniform hazard spectra for the PORTS reservation area. In the combination, the LLNL and EPRI results were given equal weight to obtain an overall representation of the seismic hazard and its associated uncertainty. Details of the process are described in Risk Engineering, Inc. (Risk Engineering, 1992). Resulting mean hazard and fractile curves are presented for both soil and rock site conditions, as well as the median uniform hazard spectra for the 2.0×10^{-3}, 1.0×10^{-3}, and 2.0×10^{-4} exceedance probabilities. Probabilistic studies were also performed by ECS to obtain the uniform hazard spectra for the PORTS reservation area for the 1.0×10^{-4} exceedance probability. The mean peak ground acceleration (rock site) corresponding to the 1.0×10^{-4} exceedance probability was estimated to be 0.20g (ECS, 2006). The mean peak ground acceleration calculated from the USGS 2002 study (Frankel, 2002) for the site vicinity was estimated to be 0.19g. The Risk Engineering study (1992) equally weighted the results of the EPRI and LLNL studies to obtain combined hazard estimates for the Portsmouth facility. At the 1.0 x 10-4 hazard level, two peak ground accelerations were obtained: 0.13g and 0.29g. The larger value of 0.29 g is the combined EPRI/LLNL result when all five of LLNL expert's inputs were considered. This larger value is caused by one model selected by ground-motion Expert 5, predicted substantially higher ground motions than the others for peak ground acceleration (and response spectrum amplitudes). This model (referred to as G16-A3) was given zero weight by four of the LLNL Experts (and 100 percent by the fifth). When excluding this outlier (model G-16-A3) by considering only 4 of the LLNL Expert's inputs, the estimated ground acceleration is 0.13 g. The Risk Engineering (1992) report further notes that model G16-A3 severely over-estimates ground motions in the central and eastern United States.

The mean peak ground accelerations (soil site) corresponding to the 2.0×10^{-3}, 1.0×10^{-3}, and 2.0×10^{-4} exceedance probabilities from the combined LLNL and EPRI results were estimated to be 0.10g, 0.15g, and 0.19g.

The methodology used to combine the LLNL and EPRI studies is acceptable. The choice of equal weight is justified because both studies constitute expert elicitation that incorporates a diverse set of scientific results and opinion. NRC staff previously accepted the LLNL and EPRI data, seismic sources, seismic hazard methods, and results (NRC, 1997) for sites in the central and eastern United States. Thus, staff concludes that using the combined LLNL and EPRI hazard results, including the 1,000-year return period 0.15g peak ground acceleration is technically sound. In addition, the staff concludes that using the ECS hazard results, including the 10,000-year return period (rock site) 0.20g peak ground acceleration, is technically sound. As such, the information about the seismic hazard calculation presented in the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b) is acceptable and it demonstrates compliance with regulatory requirements in 10 CFR 70.65(b)(1).

1.3.3.5.1.4 Development of Site-Specific Spectra

Site-specific design spectra were determined following the procedures in the DOE–STD–1024–92 (DOE, 1992) and described in detail in Risk Engineering, Inc. (Risk

Engineering, 1992). In particular, the procedures involve deaggregating the hazard to obtain the two dominant magnitudes and distance pairs that control the peak ground acceleration and maximum spectral velocity. Deterministic response spectral shapes associated with these two magnitude-distance pairs are then calculated and scaled. A single spectrum is created that envelopes the two calculated response spectra. In the Risk Engineering, Inc. (Risk Engineering, 1992) study, the rock conditions hazard was used to obtain the response spectra for hard rock conditions. The resulting design spectra were then transformed to the design spectra for soil conditions using a site-specific, soil-response analysis, "Site-Specific Earthquake Response Analysis for Portsmouth Gaseous Diffusion Plant, Portsmouth, Ohio" (Sykora, 1993). The design spectra for soil are documented in the redacted ISA Summary (USEC, 2003b) for the American Centrifuge Lead Cascade Facility. Additional evaluations were made to address the uncertainty of the low-frequency range (2.5 Hertz and less) of the response spectra and are documented in ES/CNPE–1995/2, "Seismic Hazard Criteria for the Oak Ridge, Tennessee, Paducah, Kentucky, and Portsmouth, Ohio, U.S. Department of Energy Reservations" (DOE, 1995). The response spectra provided in ES/CNPE–1995/2 account for uncertainties in the soil response spectra and provide additional seismic reserve capacity in the buildings designed in accordance with the 1978 criteria.

The general DBE for the ACP is the 1,000-year return period earthquake. According to the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b), the new structural addition to the existing X–3346 Feed and Customer Services Building, however, has a 10,000-year return DBE. Because the study by Risk Engineering, Inc. (Risk Engineering, 1992) only developed design spectra up to the 1,000-year return period, ECS developed a site-specific design response spectrum was developed by ECS for the 10,000-year return period earthquake. The details of this site-specific soil-response analysis and the development of the design spectra are documented in the ECS report (USEC, 2006e). The site-specific peak ground acceleration value for the 10,000-year return period earthquake is 0.48g. A comparison with results from an independent analysis performed for NRC staff by the Center for Nuclear Waste Regulatory Analyses using ProShake (EduPro, 2003) showed that the 10,000-year design response spectrum developed by ECS envelopes these results at most structural frequencies, with the exception of the ~5 to 10 Hz structural frequency range. Notably, the design spectra are exceeded by up to ~0.3g at frequencies between ~5 to 10 Hz. The design spectrum in this frequency range is 0.85g. However, the "International Building Code" (IBC, 2003) allows the design spectral response acceleration to be two thirds of the site-specific response spectrum at any frequency. The vertical response spectra for the 1,000-year and 10,000-year return period earthquakes are two-thirds of the horizontal response spectra.

The methodology and results used to develop site-specific spectra are acceptable. These methodologies follow modern practice. Thus, staff concludes that the site-specific hazard and response spectra are technically sound. As such, the information presented in the LA (USEC, 2006c) is acceptable and it demonstrates compliance with regulatory requirements in 10 CFR 70.65(b)(1).

1.3.3.5.1.5 Surface Faulting

There is no geologic, geophysical, or seismological evidence of active surface faulting at or nearby the PORTS reservation, as stated in Section 1.3.6.6 of the LA (USEC, 2006c). Data listed in the ISA Summary (USEC, 2006b) also suggest that the Paleozoic bedrock beneath the site is unfaulted. Therefore, surface faulting is not considered a credible disruptive event for the

proposed ACP.

1.3.3.5.2 Slope Stability

Slope stability is discussed in Section 1.3.6.4, of the LA (USEC, 2006c) and Section 1.6.4 of the ISA Summary (USEC, 2006b). As indicated, the slopes at the proposed ACP site have a horizontal-to-vertical ratio of greater than 3. These low-inclination slopes have a static safety factor of greater than 2.0 and a dynamic safety factor greater than 1.5, for a peak ground acceleration of 0.21g (USEC, 2006c).

In review of the safety factors reported in the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b) for the slopes at the site, the staff concludes that these slopes pose no threat of instability. The staff's site visit conducted as part of the Lead Cascade LA (USEC, 2003c) review further confirmed that the area at the proposed ACP site is relatively flat, as indicated in Section 1.2.3.E.ii of the SER for the Lead Cascade (NRC, 2004b). Consequently, slope stability is not a safety concern for this proposed facility.

1.3.3.5.3 Liquefaction

Liquefaction potential of soils beneath the proposed ACP is discussed in Sections 1.3.6.4 and 1.3.6.7 of the LA (USEC, 2006c) and Sections 1.6.4 and 1.6.7 of the ISA Summary (USEC, 2006b). The applicant states in the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b) that most soils at the proposed ACP site are cohesive and exhibit a low potential for liquefaction.

An extensive geotechnical investigation was conducted in the surrounding area, including the proposed ACP site, by the Law Engineering Testing Company, "Geotechnical Investigation, Gas Centrifuge Enrichment Plant, Portsmouth, Ohio" (Law Engineering, 1978). The investigation results show that the soils beneath the proposed ACP site consist mainly of inorganic silts and clays with a plastic limit of 50 percent or less. These soils contain 28-percent fines content (smaller than #200 sieve) or more with an average of 43 percent. The thickness of the soil layers ranges from approximately 9 to 12 m (30 to 40 ft), and the groundwater level is approximately 3 to 4.6 m (10 to 15 ft) below the ground. The standard penetration test results indicate that the majority of the soil samples tested beneath the proposed ACP site has a blow count of more than 20.

Based on characteristics of the soils at the site, Law Engineering Testing Company (Law Engineering, 1978) and the LA (USEC, 2006c) concluded that the soils at the site have little potential for liquefaction.

The liquefaction potential for soils also can be estimated using an approach suggested by the National Center for Earthquake Engineering Research, "Proceedings of the National Center for Earthquake Engineering Research Workshop on Evaluation of Liquefaction Resistance of Soils" (NCEER, 1997). This approach presented cyclic resistance ratio (CRR) curves for various fines contents in a cyclic stress ratio (CSR)-blow count diagram (Figure 1-1). These curves are developed based on field observations for earthquakes with Richter magnitudes near 7.5. Note that the long dash line shown in the figure was the recommended extension of the curve with 5-percent fine content to intersect the Y-axis of the figure by the Workshop Committee for the National Center for Earthquake Engineering Research.

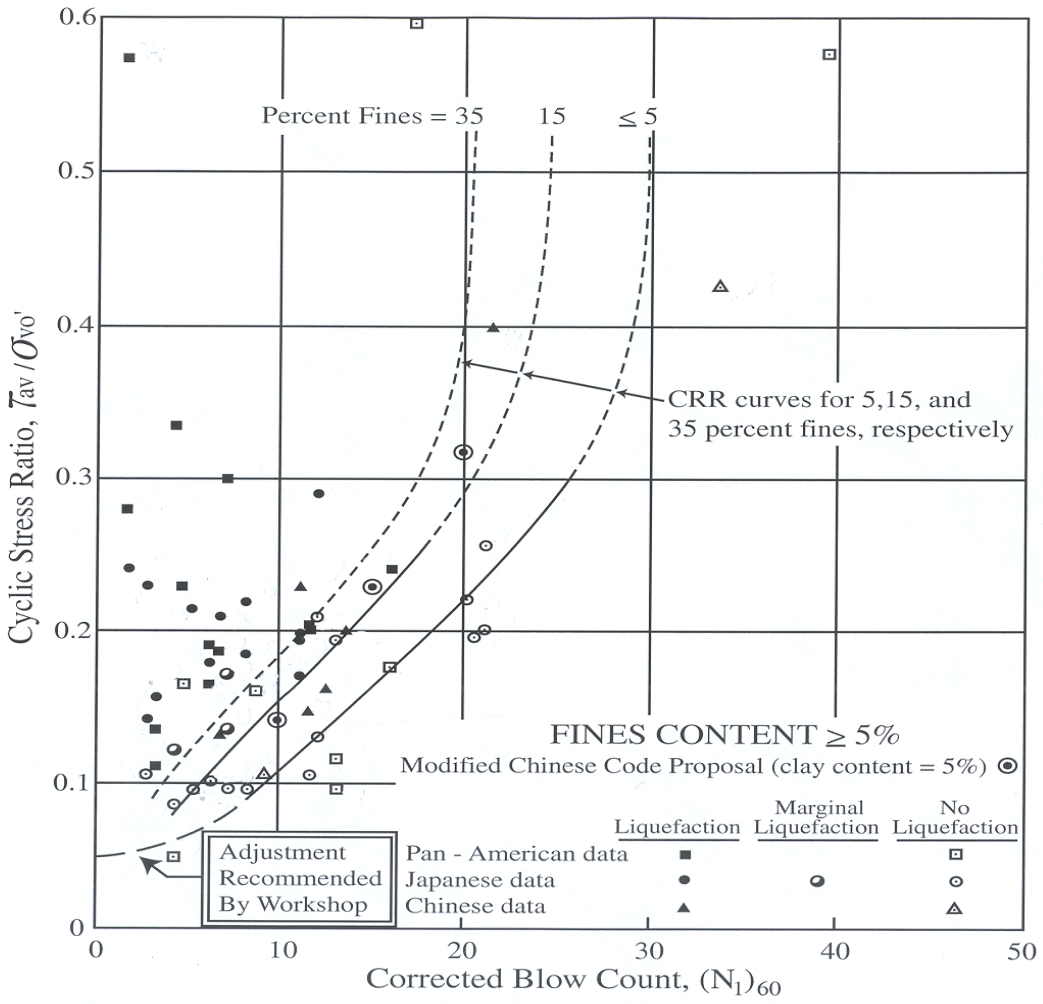

Figure 1-1 Simplified Base Curve Recommended for Calculation of CRR From the Standard Penetration Test Results (Modified from National Center for Earthquake Engineering Research, 1997)

For earthquakes of different magnitudes, the CRR curves shown in Figure 1 must be scaled up or down on the diagram. The suggested magnitude scaling factors can be found in NCEER (NCEER, 1997). According to the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b), the general DBE for the proposed ACP except the X–3346 building Customer Service Area has a Richter magnitude of 5.25 with a horizontal peak ground acceleration of 0.15g. The magnitude scaling factor for this earthquake magnitude is at least 1.43. The magnitude scaling factor of 1.43 is suggested for an earthquake of magnitude 5.5 and may be considered as a bounding value compared with the values suggested by other researchers (Table 3, NCEER, 1997). Applying this scaling factor, the CRR curves in Figure 1-1 represent a safety factor of at least 1.43 for earthquakes of magnitude 5.25.

The CSR can be calculated using the following equation (NCEER, 1997)

$$CSR = 0.65\left(\frac{a_{max}}{g}\right)\left(\frac{s_t}{s_e}\right)r_d \qquad \text{Eq. 1}$$

where a_{max} is the horizontal peak ground acceleration at ground surface, g is the acceleration of gravity, σ_t is the total vertical overburden stress, σ_e is the effective vertical overburden stress, and r_d is a stress reduction factor. The stress reduction factor, r_d, is overburden depth-dependent and can be estimated roughly (NCEER, 1997) by:

$$r_d = 1.0 - 0.00765z \quad \text{for } z \le 9.15\text{m}$$
$$r_d = 1.174 - 0.0267z \quad \text{for } 9.15\text{m} < z \le 23\text{m} \qquad \text{Eq. 2}$$

where z is the overburden depth in meters.

Using Equations (1) and (2) and assuming the groundwater level is 3 m [10 ft] below ground, for soils with a density of 1.76×10^3 kg/m^3 (110 pounds per cubic foot) (Law Engineering, 1978), the staff determined that the CSRs for soils at overburden depths of 6.1, 9.1, and 12.2 m [20, 30, and 40 ft] are approximately 0.13, 0.146, and 0.144, respectively. The blow counts from a majority of standard penetration test results are more than 20, thus the aforementioned CSRs fall in a region on the right side of the CRR curves, indicating there is no liquefaction potential for the soils at the site. Considering the smallest blow count value, 8, experienced for one of the soil sample tests, the aforementioned CSRs still will fall within the no-liquefaction region defined by the cyclic resistance ratio curve with 35-percent fines content.

For the DBE for the X–3346 building Customer Service Area (a 10,000-year return period earthquake) with a horizontal peak ground acceleration of 0.48g, the corresponding CSRs are larger. However, CSRs for the majority of standard penetration test locations, after considering the scale factor, should still fall within the no-liquefaction region defined by the CRR curve with 35-percent fines content because the blow counts from a majority of standard penetration test results are more than 20.

The staff reviewed the geotechnical investigation report, conducted an independent assessment, and concludes that liquefaction of soils at the site is not a safety concern for the proposed ACP.

1.3.3.5.4 Settlement

Settlement of foundations for the proposed ACP is discussed briefly in Section 1.3.6.4 of the LA (USEC, 2006c) and Section 1.6.4 of the ISA Summary (USEC, 2006b). As stated in the LA (USEC, 2006c) and ISA Summary (USEC, 2006b), the predicted total settlement of foundations is expected to be less than 5.1 cm (2 in).

Staff selectively reviewed the design and as-built structural drawings of the X–3001 Process Building and finds the foundations were designed and constructed with individual spread footings and piers to support building columns (DOE, 1982). Differential settlements between footings could result because of soil variability. In its geotechnical investigation report, Law Engineering Testing Company provides the average anticipated settlement of footings for the

X–3001 Process Building based on the soil property data determined for the site (Figure 10–7A, Law Engineering, 1978). The geotechnical investigation report also estimated a ±25-percent variation in soil properties within a 15.2-m (50-ft) radius of any given point of the foundation area based on soil property data obtained from beneath the proposed ACP site.

Assuming a column load of 181×10^3 kg (400×10^3 lbs) and a bearing pressure of 287.3 kPa [6,000 psf], the settlement of the footing can be estimated to be approximately 1.93 cm (0.76 in) (from Figure 10–7A, Law Engineering, 1978). Thus, a differential settlement of 0.97 cm (0.38 in) {50 percent of the settlement, 1.93 cm (0.76 in), of a footing} can be estimated within a 15.2-m (50-ft) distance.

A separate settlement calculation was performed by Pro2Serve (Pro2Serve, 2003) at the request of USEC. The result shows an approximate 1.27 cm (0.5 in) of footing settlement and a differential settlement of 0.64 cm (0.25 in) within a 15.2-m (50-ft) distance, assuming a column load of 181×10^3 kg (400×10^3 lbs) and a bearing pressure of 287.3 kPa (6,000 psf). This estimated settlement value is 0.66 cm (0.26 in) smaller than that estimated from the Law Engineering Testing Company site investigation report. The difference may be largely because of the variation in the soil property data set used. The settlement estimate suggested by Pro2Serve (Pro2Serve, 2003) uses the data of the soils located in the immediate vicinity of the X–3001 Process Building, whereas Law Engineering Testing Company uses the soil data collected for a wider region, including those soils below the Process Building. The methods used for settlement calculations also may be responsible for the difference. The Law Engineering Testing Company used the Westergaard (Westergaard, 1938) stress distribution method. Pro2Serve (Pro2Serve, 2003) used two methods—the Westergaard stress distribution method to determine the stress increase with depth beneath the footing and the Bowles (Bowles, 1996) method for settlement calculation. The estimate from Law Engineering Testing Company is relatively more conservative for the X–3001 Process Building.

By conservatively assuming a ±50-percent variation in soil properties within a 30.5-m [100-ft] distance, the differential settlement between footings is approximately 1.93 cm [0.76 in] based on Law Engineering Testing Company and 1.27 cm [0.5 in] based on Pro2Serve (Pro2Serve, 2003).

The distance of two adjacent columns for the Process Building is 31.7 m [104 ft] along the east-west direction and 6.1 m [20 ft] along the north-south direction. The design basis differential settlement used for the design and construction of the X–3001 Process Building is 2.54 cm [1 in] (Fluor, 1978a, 1978b) irrespective of the distance between two adjacent footings. This design basis differential settlement is larger than those estimated by Law Engineering Testing Company and Pro2Serve for a column load of 181×10^3 kg [400×10^3 lbs] and a bearing pressure of 287.3 kPa [6,000 psf]. Pro2Serve (Pro2Serve, 2003) also calculated the footing settlement using the actual footing size from the as-built structural drawings and the actual design footing loads. The estimated footing differential settlement within a 30.5-m [100-ft] distance is approximately 0.76 cm [0.3 in]. This value is substantially smaller than the design basis value.

Staff reviewed the information presented for calculating differential settlements for the X–3001 Process Building and finds that the differential settlements determined between adjacent columns of the Process Building are based on site-specific soil data and the methodologies used are acceptable. The staff also finds that the design basis differential settlement used for

the design and construction of the Process Building is acceptable because it bounds the estimated differential settlements. Because the review of the development of the design basis differential settlement for the X–3001 Process Building was conducted as a sample case, the staff has found reasonable assurance that the applicant used the same process to determine the design basis differential settlements for other existing buildings.

The design basis differential settlements for the new buildings of the proposed ACP are not available, and development of these design bases will be performed in parallel with the design of the new buildings of the proposed ACP (USEC, 2005). The staff finds this approach acceptable and will assess the acceptability of the design basis differential settlements, including the methods used once the design basis differential settlements become available.

1.3.3.5.5 Geology Conclusion

The applicant provided a summary description of the geology, including seismicity, for the area and provides earthquake accelerations for the site associated with 1000-year and 10,000-year earthquakes. The applicant's descriptions are consistent with the more detailed information in the ISA Summary (USEC, 2006b), the ER (USEC, 2003a), and the EP (USEC, 2006a). The information provided by the applicant provided is consistent with the guidance in Sections 1.3.4.3(4) and 1.3.4.3(5) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.3.4 Evaluation Findings

The staff has reviewed the site description for the proposed ACP uranium enrichment facility according to Section 1.3 of the Standard Review Plan (NRC, 2002). The applicant has adequately described and summarized general information pertaining to: (1) the site geography, including its location relative to prominent natural and man-made features such as mountains, rivers, airports, population centers, schools, and commercial and manufacturing facilities; (2) population information on the basis of the most current available census data to show population distribution as a function of distance from the facility; (3) meteorology, hydrology, and geology for the site; and (4) applicable design basis events. The reviewer verified that the site description is consistent with the information used as a basis for the ER (USEC, 2003a), the EP (USEC, 2006a), and ISA Summary (USEC, 2006b); and that it demonstrates compliance with regulatory requirements in 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, and 10 CFR 70.65(b)(1).

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2.0 ORGANIZATION AND ADMINISTRATION

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's organization and administration is to evaluate whether the application describes proposed management policies that provide reasonable assurance that the licensee plans, implements, and controls site activities in a manner that ensures the safety of workers and the public, and protects the environment. The review also ensures that the applicant has identified and provided adequate qualification descriptions for key management positions.

2.1 REGULATORY REQUIREMENTS

Related requirements in 10 CFR 70.22 and 70.23 pertain to the establishment of a management system and administrative procedures for the effective implementation of health, safety, and environment (HS&E) functions. Effectively implementing these functions will better ensure adequate safety for workers and the public, and protection of the environment.

2.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the organization and administration section of the license application (LA) (USEC, 2006b) is contained in Chapter 2 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). Section 2.3 of NUREG-1520, "Areas of Review," includes areas of review for both new facility applications and applications for modifications of existing facilities. Because the American Centrifuge Plant (ACP) is a new facility, the areas of review for existing facilities are not applicable. Similarly, Section 2.4.3 of NUREG-1520, "Regulatory Acceptance Criteria," lists acceptance criteria for both new facilities and existing facilities. Only the Regulatory Acceptance Criteria for new facilities are applicable to the LA (USEC, 2006b). The acceptance criteria applicable to this review are contained in Section 2.4.3 of NUREG-1520 (NRC, 2002).

2.3 STAFF REVIEW AND ANALYSIS

In Section 2.0 of the applicant's LA (USEC, 2006b), the applicant commits to the following policy:

> USEC is responsible for safe operation of the ACP and is committed to conducting operations in a manner that protects the health and safety of workers and the public; protects the environment; provides for the common defense and security; and is in compliance with applicable local, state, and federal laws and regulations.

In Section 2.0 of the LA (USEC, 2006b), the applicant further states that it will be responsible for the design, quality assurance (QA), refurbishment/construction, testing, start-up, operation, maintenance, and decommissioning of the ACP and that qualified individuals will ensure a smooth transition from refurbishment/construction activities to plant operations. The Engineering Manager has the responsibility for construction management and coordination with the contractor(s). The operations organization will be responsible for the safe operation of the ACP. Programs and staff organizations will be established to cover the HS&E, safeguards, security, and QA areas. These programs and organizations will be provided with sufficient

resources to support safe operation of the ACP.

2.3.1 Organization

In Section 2.1 of the LA (USEC, 2006b), the applicant describes the organizational commitments, relationships, responsibilities, and authorities for the overall management system to assure the protection of the health and safety of the workers and the public, protection of the environment, and to provide for the common defense and security. This section includes the qualifications, functions, responsibilities, and authorities of the positions in the organizations assigned functions related to environmental, health, safety, safeguards, security, and QA during all stages of the project, from design through refurbishment/construction, start-up, operation, and decommissioning. In Section 2.0 of the LA (USEC, 2006b), the applicant states that the qualifications, responsibilities, and authorities are defined in position descriptions that will be accessible to affected personnel and NRC request.

Figure 2.1-1 of the LA (USEC, 2006b) depicts the ACP organization from design through refurbishment/construction, start-up, and operation.

As stated above, the applicant has identified and provided a description of the proposed project organization that would be responsible for managing the design, construction, and operation of the proposed facility. The applicant has also provided organization charts. The proposed organization provides an acceptable management system for ensuring that the design, construction, and operation of the facility will meet NRC regulatory requirements. The information provided by the applicant meets the guidance in Section 2.4.3(1) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

2.3.2 Organizational Responsibilities and Qualifications

In Section 2.1 of the LA (USEC, 2006b), the applicant provided information concerning the minimum qualifications, functions and responsibilities for key staff positions. Personnel responsible for managing the design, refurbishment/construction, and operation of the plant will be required to have the substantive breadth and level of experience to successfully execute their responsibilities. The personnel filling these key staff positions will be located at the plant and will be available as necessary. Alternates will be designated in writing in accordance with procedural requirements to fulfill the responsibilities and authorities of these personnel during their absence from the facility.

According to Section 2.1 of the LA (USEC, 2006b), equivalent technical experience means the substitution of 2 years of nuclear industry experience for each year of college up to a total of 3 years. Additionally, 30 semester hours or 45-quarter hours from an accredited college or university may be substituted for the remaining 1 year of baccalaureate education. Individuals who do not possess the formal educational requirements specified in this section or do not meet the equivalent technical experience defined above will not be automatically eliminated where other factors provide sufficient demonstration of their abilities to fulfill the duties of a specific position. These other factors must clearly demonstrate proficiency in the technical area for which the position will be responsible (e.g, a license or certification, documented completion of relevant training, or previous experience in the same position at another facility). These factors will be evaluated on a case-by-case basis, documented, and approved by the Director, American Centrifuge Plant.

The Vice President, American Centrifuge, located at USEC Inc. (USEC) Headquarters, reports to the Senior Vice President. The Vice President, American Centrifuge, has overall responsibility for safe operation of the ACP and has shut-down and stop-work authority. If shut down and stop work authority is exercised, the Vice President, American Centrifuge, must concur with restart of shutdown operations. The Vice President, American Centrifuge, has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, 6 years nuclear experience, and 10 years of management experience (which may be concurrent with the nuclear experience). The Vice President, American Centrifuge, will be appointed by the USEC Board of Directors.

The Director, Regulatory and Quality Assurance, is be located at USEC Headquarters. The individual holding this position will be appointed by and reports to the Vice President, American Centrifuge. The individual holding this position will be responsible for the management of regulatory and QA functions as well as the ACP policy system. The individual holding this position will be the primary day-to-day interface with the NRC and has overall responsibility for the management of activities related to license requirements for the ACP. This individual will be independent from production, plant operating cost, and production schedule concerns, and has the authority to stop work activities if there will be a failure to adhere to regulatory requirements. If such authority is exercised, the Director, Regulatory and Quality Assurance, must concur with restart of shutdown operations. The individual holding this position must have, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, six years nuclear experience, and 6 years management experience. Management experience may be concurrent with the nuclear experience.

The Regulatory Manager, who reports to the Director, Regulatory and Quality Assurance, has responsibility for regulatory oversight functions, environmental compliance, and commitment management. This individual, located at the ACP, will be responsible for managing the plant change process, implementing the Corrective Action Program, ensuring incident investigations are performed, and providing management with data to assure that corrective actions and commitments are properly addressed and managed. The Regulatory Manager will be appointed by the Director, Regulatory and Quality Assurance, with concurrence from the Director, American Centrifuge Plant. The Regulatory Manager has shutdown and stop work authority in any part of the ACP where activities are not being conducted in accordance with applicable regulatory requirements. If such authority is exercised, the Regulatory Manager must concur with restart of shutdown operations. The Regulatory Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and 4 years of experience in the nuclear industry.

The QA Manager, who reports to the Director, Regulatory and Quality Assurance, has responsibility for oversight of procurement, refurbishment, construction, start-up, and plant operations to ensure adequate protection of the health and safety of workers and the public; to ensure compliance with safety, safeguards, and quality requirements; and to ensure implementation of the Quality Assurance Program Description (QAPD) and policies, procedures, and management expectations for the ACP. The QA Manager, located at the ACP, interacts directly with the Vice President, American Centrifuge, other managers, and key ACP personnel and participates in evaluations or discussions related to safety, safeguards, and quality. The QA Manager will be appointed by the Director, Regulatory and Quality Assurance, with concurrence from the Vice President, American Centrifuge. The QA Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical

experience, 4 years of experience in the nuclear industry, and 4 years of management experience in QA, nuclear safety oversight, engineering and technical support, or regulatory affairs. Management experience may be concurrent with nuclear experience.

The Director, American Centrifuge Plant reports to the Vice President, American Centrifuge, and is located at the ACP. The Director, American Centrifuge Plant, will be responsible for the overall safe operation and maintenance of the ACP. These responsibilities include refurbishment/construction, initial start-up, testing operation, training, procedures, engineering, as well as occupational, environmental, and nuclear safety. This individual has the primary responsibility for the interface with NRC inspection personnel on matters of regulatory compliance, and may delegate the responsibility for routine interface with NRC inspection personnel to the Regulatory Manager. The Director, American Centrifuge Plant, will be appointed by the Vice President, American Centrifuge. The Director, American Centrifuge Plant has shutdown and stop work authority for the ACP, and if such authority is exercised, must concur with restart of shutdown operations. The Director, American Centrifuge Plant, must obtain concurrence of the Vice President, American Centrifuge, for restart of any operations that were directed to be shutdown by the Quality Assurance Manager or the Director, Regulatory and Quality Assurance. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, 6 years of nuclear experience, and 6 years of management experience. Management experience may be concurrent with nuclear experience.

The Plant Support Manager will be responsible for Fire Safety, Health Services, Emergency Management, and (NMC&A) for the ACP. This individual reports to the Director, American Centrifuge Plant, and, in the absence of the Director, American Centrifuge Plant, the Plant Support Manager may be delegated the responsibilities and authorities of the Director, American Centrifuge Plant. The Director, American Centrifuge Plant appoints the Plant Support Manager, with concurrence from the Vice President, American Centrifuge. The Plant Support Manager has shutdown and stop work authority in any part of the ACP where activities are not being conducted in accordance with applicable regulatory requirements for which the Plant Support Manager has responsibility. If such authority is exercised, the Plant Support Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Fire Safety Manager will be responsible for fire protection services including the interpretation and application of fire codes and standards, and emergency management. This individual reports to and will be appointed by the Plant Support Manager, with concurrence from the Director, American Centrifuge Plant. The Fire Safety Manager has shutdown and stop work authority for activities at the ACP not being conducted in accordance with applicable fire protection requirements. If such authority is exercised, the Fire Safety Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree or equivalent technical experience, four years of fire protection experience, and six months of nuclear experience.

The Nuclear Materials Control and Accountability (NMC&A) Manager will be responsible for ensuring the effective implementation of the NMC&A program. This individual reports to and will be appointed by the Plant Support Manager, with the concurrence of the Director, American Centrifuge Plant. The NMC&A Manager has shutdown and stop work authority for activities at

the ACP not being conducted in accordance with NMC&A requirements. If such authority is exercised, the NMC&A Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree or equivalent technical experience, and 4 years of NMC&A experience.

The Engineering Manager will be responsible for engineering activities in support of operations including plant projects, system engineering, procurement, construction management and construction engineering, records management, document control, the Nuclear Criticality Safety (NCS) program, maintaining the ACP Integrated Safety Analysis (ISA), and management of the design change process for the ACP. The Engineering Manager reports to and will be appointed by the Director, American Centrifuge Plant, with concurrence from the Vice President, American Centrifuge. In the absence of the Director, American Centrifuge Plant, the Engineering Manager may be delegated the responsibilities and authorities of the Director, American Centrifuge Plant. The Engineering Manager has shutdown and stop work authority for any activity that poses a nuclear safety or criticality concern; or any activity that would be or is in violation of the ACP's licensing or design basis, or the assumptions or evaluations contained in the ISA Summary (USEC, 2006a). If such authority is exercised, the Engineering Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and 4 years of nuclear experience.

The Nuclear Safety Manager will be responsible for developing and implementing the safety analysis program for the ACP. These duties include technical oversight of safety analysis, safety analysis training, review of procedures involving fissile material operations, and assessments of program implementation. The Nuclear Safety Manager is also responsible for procurement engineering and configuration management. The Nuclear Safety Manager has direct access to the Director, American Centrifuge Plant concerning nuclear safety matters and has shutdown and stop work authority for any activity that would be or is in violation of the ACP's licensing or design basis, or the assumptions or evaluations contained in the ISA Summary (USEC, 2006a). If such authority is exercised, the Nuclear Safety Manager must concur with restart of shutdown operations. The Nuclear Safety Manager reports to and will be appointed by the Engineering Manager with the concurrence of the Director, American Centrifuge Plant. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and 4 years of nuclear experience, including six months of experience at a uranium processing plant.

The NCS Manager will be responsible for the management of NCS functions and administration of the NCS program. The NCS Manager reports to and will be appointed by the Nuclear Safety Manager with the concurrence of the Engineering Manager. The NCS Manager has stop work authority for any activity that could cause a NCS concern. If such authority is exercised, the NCS Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience, including 6 months of experience at a uranium processing facility where NCS was practiced.

The Manager, Enrichment Operations, will be responsible for the day-to-day production activities at the ACP including production support, operations, and maintenance. The Manager, Enrichment Operations, reports to and will be appointed by the Director, American Centrifuge Plant, with concurrence from the Vice President, American Centrifuge. In the absence of the

Director, American Centrifuge Plant, the Manager, Enrichment Operations, may be delegated the responsibilities and authorities of the Director, American Centrifuge Plant. The Manager, Enrichment Operations, has shutdown and stop work authority in any part of the ACP where activities are not being conducted in accordance with applicable regulatory requirements. If such authority is exercised, the Manager, Enrichment Operations, must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and 4 years of nuclear experience.

The Production Support Manager will be responsible for industrial safety, industrial hygiene, chemical safety, and the Radiation Protection Program. Additional responsibilities include waste management, environmental survey, training, and procedures. The Production Support Manager reports to and will be appointed by the Manager, Enrichment Operations, with concurrence from the Director, American Centrifuge Plant. In the absence of the Manager, Enrichment Operations, the Production Support Manager may be delegated the responsibilities and authorities of the Manager, Enrichment Operations. The Production Support Manager has shutdown and stop work authority in any part of the operation for which he/she has responsibility. If such authority is exercised, the Production Support Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and 4 years of nuclear experience.

The Radiation Protection Manager (RPM) will be responsible for the establishment and implementation of the ACP Radiation Protection Program. The RPM has the authority to deny access to radiological areas by personnel who do not adhere to radiological protection requirements. The RPM reports to and will be appointed by the Production Support Manager, with the concurrence of the Manager, Enrichment Operations. The RPM has direct access to the Director, American Centrifuge Plant, and the Vice President, American Centrifuge, for radiation protection matters. The RPM has shutdown and stop work authority for activities not being conducted in accordance with radiation protection requirements and policies. If such authority is exercised, the RPM must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering, health physics, radiation protection, or the physical sciences or equivalent technical experience, and four years of experience in radiation protection, including 6 months at a uranium processing plant.

The Training Manager will be responsible for the development and implementation of the technical and qualification training program, the development and implementation of the procedures program, and for preparation, presentation, and documentation of employee orientations. The Training Manager reports to and will be appointed by the Production Support Manager, with concurrence from the Manager, Enrichment Operations. The Training Manager has shutdown and stop work authority in any part of the operation for which he/she has responsibility. If such authority is exercised, the Training Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and 4 years of nuclear experience.

The Operations Manager will be responsible for enrichment operations, feed and withdrawal operations, utilities, production management, shift operations, packaging and transportation,

and repair and assembly of centrifuge machines. These responsibilities include: ensuring the correct and safe operation of the enrichment process; proper receipt, storage, handling, and on-site transportation of uranium hexafluoride (UF_6); and providing chemical cleaning and decontamination services. In the absence of the Manager, Enrichment Operations, the Operations Manager may be delegated the responsibilities and authorities of the Manager, Enrichment Operations. The Operations Manager reports to and will be appointed by the Manager, Enrichment Operations, with concurrence from the Director, American Centrifuge Plant. The Operations Manager has shutdown and stop work authority in any part of the operation for which he/she has responsibility. If such authority is exercised, the Operations Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience, including 6 months at a uranium processing plant.

Operations Supervisors report to and will be appointed by the Operations Manager, with the concurrence of the Manager, Enrichment Operations. The Operations Supervisor will be the senior manager on shift, and there will be one Operations Supervisor assigned to each shift at the ACP. In this capacity, the Operations Supervisor represents the Director, American Centrifuge Plant. This individual has the responsibility and authority to make decisions, including shutdown and stop work authority, in order to place the plant in a safe condition. During emergencies, the Operations Supervisors will be responsible for dissemination of information regarding plant activities to the incident commander, and making event notifications to regulatory agencies. Operations Supervisors will be also responsible for providing operational support for centrifuge machine repair, assembly, transport, installation, pump down, testing, and start-up. The minimum qualifications for this position will be a high school diploma or satisfactory completion of the General Educational Development test, and 3 years of experience in operations, maintenance, or engineering at an industrial, chemical, or nuclear plant. Operations Supervisors must also have 1 year of supervisory experience or have completed a supervisory training course.

The Maintenance Manager will be responsible for the safe performance of preventive and corrective maintenance and support services on facilities and equipment, with the exception of centrifuge machines. These responsibilities include: maintaining the relevant logs and records; work planning/control to initiate, evaluate, and prioritize maintenance work; and coordinating shop maintenance. In the absence of the Manager, Enrichment Operations, the Maintenance Manager may be delegated the responsibilities and authorities of the Manager, Enrichment Operations. The Maintenance Manager reports to and will be appointed by the Manager, Enrichment Operations, with concurrence from the Director, American Centrifuge Plant. The Maintenance Manager has shutdown and stop work authority in any part of the operation for which he/she has responsibility. If such authority is exercised, the Maintenance Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and 4 years of nuclear experience.

Maintenance Supervisors report to and will be appointed by the Maintenance Manager. Maintenance Supervisors will be responsible for supervising the maintenance of mechanical equipment (pumps and valves), electrical equipment, electronic and pneumatic instrumentation and controls, computers, and programmable controllers. The minimum qualifications for this position will be a high school diploma or satisfactory completion of the General Educational

Development test, and three years of experience in operations, maintenance, or engineering at an industrial, chemical, or nuclear plant. The Maintenance Supervisors have shutdown and stop work authority in any part of the operation for which they have responsibility. Maintenance Supervisors must also have 1 year of supervisory experience or have completed a supervisory training course.

The applicant briefly discusses the shift crew composition at the ACP in Section 2.1.3.3.4 of the LA (USEC, 2006b). The minimum operating shift crew consists of an Operations Supervisor, a Radiation Protection/Industrial Hygiene technician, and one operations technician per process building. Other personnel, such as NCS staff, will be available on an as needed basis.

Section 2.1.4 of the LA (USEC, 2006b) describes the responsibilities and qualifications of the Corporate Security Director. The Corporate Security Director reports to the Senior Vice President, Human Resources and Administration and will be located at USEC Headquarters. The Corporate Security Director will be responsible for the strategic direction of security operations and programs, including physical, personnel, and information security. The Corporate Security Director will be appointed by the Senior Vice President, Human Resources and Administration. The minimum qualifications for this position will be a bachelor's degree or equivalent technical experience, 6 years of security experience, and 6 years of management experience. Management experience may be concurrent with security experience.

The Security Manager, located at the ACP, will be responsible for the ACP safeguards and security services. The Security Manager reports to and will be appointed by the Corporate Security Director, with the concurrence of the Director, American Centrifuge Plant. This individual has direct access to the Director, American Centrifuge Plant concerning all security matters and has shutdown and stop work authority for activities not being conducted in accordance with applicable security requirements. If such authority is exercised, the Security Manager must concur with restart of shutdown operations. The minimum qualifications for this position will be a bachelor's degree or equivalent technical experience, and 4 years of security experience.

As presented above, the applicant identified the responsibilities, qualifications, and authorities of the key personnel responsible for managing the design, construction, and operations of the proposed facility and for health, safety, and engineering responsibilities. These responsibilities, qualifications, and authorities are clearly defined and sufficient to ensure that competent management staff with sufficient experience are in place. In addition, the applicant committed to having written position descriptions that will be available to all affected personnel and to the NRC, upon request. The information applicant provided meets the guidance in Section 2.4.3(3) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

2.3.3 Management Control

Section 2.2 of the LA (USEC, 2006b) describes the management measures with associated policies, administrative procedures, and management controls to ensure that the ACP equipment, facilities and procedures, the staff (including training and qualifications), and the programs provide for the protection of the health and safety of workers and the public, protection of the environment, and for common defense and security.

Organizations having environmental, health and safety, safeguards, security, and QA functions

are independent from the operations organization providing separate and independent lines of communication. Organizations having engineering, safety and health, environmental, security, safeguards, and operations responsibilities have clear and well-defined lines of communication and authority.

Activities that will be essential for effective implementation of the environmental, safety, and health functions will be documented in approved, written procedures, prepared in compliance with a document control program.

The applicant's commitment tracking and corrective action programs will be integrated to prioritize ACP actions consistent with their safety and safeguards significance. Any person working in the facility may report potentially unsafe conditions or activities by submitting a condition notification. Reported concerns will be investigated, assessed, and resolved as described in Section 11.6 of the LA (USEC, 2006b).

Where safety, security or safeguards might be adversely impacted by cost or schedule considerations, it will be the policy of the applicant, as explained in Section 2.2 of the LA (USEC, 2006b), to subordinate cost and schedule considerations to ensure adequate treatment of safety and safeguards that will be in full compliance with applicable regulatory requirements.

The integration of ACP operations and the various programs and requirements will be accomplished through a variety of management practices, including staff meetings to discuss issues and policy implementation, review of performance indicators, review of identified events or conditions, multi-discipline reviews by the Plant Safety Review Committee (PSRC), and plant work permit systems that provide the integration in the field of various health, safety, and environmental program requirements and hazard evaluations. Additionally, oversight of the integration of various program elements will be provided by the ACP QA Organization.

The PSRC performs multi-discipline reviews of day-to-day and proposed ACP activities to ensure that these activities are and/or will be conducted in a safe manner. The PSRC advises the Director, American Centrifuge Plant, on matters related to radiation protection, nuclear safety, chemical safety, fire safety, and environmental protection. The specific membership, qualifications, meeting frequency, quorum, functions, responsibilities, and required records will be provided in a plant procedure. Auditing and oversight of PSRC activities will be the responsibility of the QA Manager.

Subcommittees may be established by the PSRC chairperson to provide assistance in conducting reviews and assessments as described in the PSRC procedure. The PSRC chairperson will approve the subcommittee procedures, membership, and member qualifications. The PSRC will maintain the overall responsibility for any required reviews.

As presented above, the applicant has provided clear, unambiguous management controls and lines of communication and authority within the organization for managing the design, construction, and operation of the facility. The proposed management controls provide an acceptable management system for ensuring that the design, construction, and operation of the facility will meet NRC regulatory requirements. The information the applicant provided meets the guidance in Section 2.4.3(2) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

2.3.4 Pre-operational Testing and Initial Start-Up

In Section 2.3 of the LA (USEC, 2006b), the applicant describes its plans and the management controls for pre-operational testing and initial start-up of the ACP, and specific plans that have been established to ensure the safe and efficient turnover, testing, and start-up of centrifuge machines, equipment, and support systems. These plans cover the transition from the refurbishment/construction phase to the operations phase.

The overall objectives of the pre-operational test program will be to ensure that the ACP facilities and systems, including the items relied on for safety (IROFS):

- Have been adequately designed and constructed;
- Meet regulatory and licensing requirements;
- Do not adversely affect workers or public health and safety; and
- Can be maintained and operated in a dependable manner so as to perform their intended function.

The refurbishment/construction contractor will be responsible for completion of as-built drawing verification, purging/flushing, cleaning, hydrostatic or pneumatic testing, system turnover, initial calibration of instrumentation in accordance with procedures, design documents, and installation specifications. As systems or portions of systems will be turned over to the applicant, acceptance testing will be performed in accordance with established schedules. The Engineering Manager will be responsible for coordination of turnover and acceptance testing.

Integrated systems testing, as a minimum, includes system or component tests required by the pertinent design codes or QAPD that were not performed by the refurbishment/construction contractor(s) before turnover to the applicant. The testing that will be performed is commensurate with the system or component's quality level and will be principally associated with IROFS, but may also include other tests on systems or components that the applicant deems appropriate for financial, reliability, or other reasons. Integrated systems tests include the testing that will be necessary to demonstrate that the facility, system, or component will be capable of performing its intended function. The Operations Manager will be responsible for coordinating the Integrated Systems Test Plan (ISTP) for the ACP. The integrated systems tests will be performed after completion of construction, flushing, hydrostatic or pneumatic testing, system turnover, and initial calibration of required instrumentation. Scheduling of the testing will be such that it generally occurs before UF_6 introduction. Other pre-operational tests, not required before UF_6 introduction, may be performed after introduction of UF_6 to the process system.

The purpose of initial start-up testing will be to ensure structures, systems, and components will perform their intended design functions in a safe and controlled manner. Examples of initial start-up tests include the leak testing, evacuation, start-up, and filling of a centrifuge machine.

As presented above, the applicant described specific plans to commission the facility's startup and operation, including the transition from the startup phase to operations phase under the direct supervision of the applicant's personnel responsible for safe operations. These organization transition plans provide adequate management and organizational controls to ensure that operations will meet NRC regulatory requirements. The information the applicant

provided meets the guidance in Section 2.4.3(4) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

2.4 EVALUATION FINDINGS

The staff has reviewed the organization and administration for the ACP in accordance with the acceptance criteria in Chapter 2 of NUREG-1520 (NRC, 2002). The staff reviewed the applicant's organization, management position summaries and qualifications, and management controls. These organizational and administrative elements describe: (1) clear responsibilities and associated resources for the design, construction, and operation of the facility, and (2) the applicant's plans for managing and operating the project. The staff has reviewed these plans and commitments and concludes that they provide reasonable assurance that an acceptable organization, administrative policies, and sufficient competent resources have been or will be established in such a manner as will allow for the safe operation of the facility.

2.5 REFERENCES

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

(USEC, 2006a) USEC Inc. (USEC). "Integrated Safety Analysis Summary for the American Centrifuge Plant in Piketon, Ohio," Revision 14, August 2006.

(USEC, 2006b) USEC Inc. (USEC). "License Application for the American Centrifuge Plant in Piketon, Ohio," Revision 18, September 2006.

3.0 INTEGRATED SAFETY ANALYSIS (ISA) AND ISA SUMMARY

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's Integrated Safety Analysis (ISA) and ISA Summary (USEC, 2006a) is to evaluate whether the application meets 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 determined whether appropriate hazards and baseline design criteria (BDC) have been addressed. The review also determined whether acceptable items relied on for safety (IROFS) including initial conditions (ICs), 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. The review also determined whether programmatic commitments to maintain the ISA and ISA Summary are acceptable.

In particular, this review considered information the applicant provided related to:

- The use of BDC for the design of the facility, in accordance with 10 CFR 70.64(a);
- Commitments regarding the applicant's safety program, including the ISA, pursuant to the requirements of 10 CFR 70.62; and
- ISA summaries submitted in accordance with 10 CFR 70.62(c)(3)(ii) and 70.65.

3.1 REGULATORY REQUIREMENTS

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

1. 10 CFR 70.62 specifies the requirement to establish and maintain a safety program, including performance of 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 BDC and facility and system design and facility layout; and
4. 10 CFR 70.65(b) describes the contents of an ISA Summary.

The regulations in 10 CFR 70.62, require an applicant to establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. The safety program is required to contain: (1) process safety information; (2) an ISA; and (3) management measures. The ISA must be conducted and maintained by the applicant and must identify the following:

- Radiological hazards related to possessing or processing licensed material;
- Chemical hazards of licensed material and hazardous chemicals produced from licensed material;
- Facility hazards that could affect the safety of licensed materials and thus present an increased radiological risk;
- Potential accident sequences caused by process deviations or other events internal to the facility and credible external events, including natural phenomena;
- Consequences and likelihood of occurrence of each potential accident sequence identified and the methods used to determine the consequences and likelihood; and
- Each IROFS identified pursuant to 10 CFR 70.61; the characteristics of its preventative, mitigative; or other safety function and the assumptions and conditions under which the item is relied to support compliance with 10 CFR 70.61.

The regulations, in 10 CFR 70.61, provide that the ISA must evaluate compliance with performance requirements. These requirements specify that the risk of each credible high-consequence event must be limited such that the likelihood of occurrence is highly unlikely, and the risk of each credible intermediate consequence event must be limited such that the likelihood of occurrence is unlikely.

The license application (LA) must include a description of the safety program under 10 CFR 70.65(a). In addition, the applicant is required to submit to NRC an ISA Summary. The Summary is required to contain:

- A general description of the site with emphasis on those factors that could affect safety;
- A general description of the facility with emphasis on those areas that could affect safety;
- 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 BDC 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 its 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 exceeded the performance requirements of 10 CFR 70.61; and

- A description of the definitions of unlikely, highly unlikely, and credible, as used in the evaluations in the ISA.

3.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the applicant's ISA and ISA Summary (USEC, 2006a) is contained in Chapter 3 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). These sections are applicable in their entirety with three exceptions. The first exception is 3.4.3.2(4)(c), which addresses criticality monitoring, because criticality monitoring is addressed in Chapter 5 of this Safety Evaluation Report (SER). The second exception is Section 3.4.3.2(5)b(i-ix), regarding process hazard analysis methods. This section provides conditions that should be met for hazard analysis methods used by the applicant if the methods are not described in NUREG-1513 (NRC, 2001). Because the methods used by the applicant (preliminary hazards analysis method and the what if/checklist method (WI/CL)) are described in NUREG-1513 (NRC, 2001), these conditions in Section 3.4.3.2(5)b(i-ix) do not have to be addressed. The third exception is various subparts in Section 3.4.3.2(9) regarding qualitative methods of defining and evaluating likelihood, because the applicant uses a quantitative method. The acceptance criteria applicable to this review are contained in Sections 3.4.3.1 and 3.4.3.2 of NUREG-1520 (NRC, 2002).

3.3 STAFF REVIEW AND ANALYSIS

This Chapter of the SER primarily documents the staff's review of the applicant's commitment to compile and maintain process safety information, the ISA methodology, and the implementation of the BDC and defense-in-depth. The appendices of this SER document the staff's review of accident sequences, IROFS, and other information that is categorized and marked as "Official Use Only - Department of Energy NOFORN" because it contains export controlled information. Appendix A of this SER contains an overall review of the ISA Summary (USEC, 2006a). A summary of the ISA results including the identification of hazards, formulation of accident sequences, selection of IROFS and management measures, and compliance with the performance requirements of 10 CFR 70.61 are included in Section A.3.1.7 of this SER.

The staff reviewed the safety program as described in the applicant's ISA Summary (USEC, 2006a) and LA (USEC, 2006b) to assess compliance with the regulatory requirements. This includes information describing the site, facility, processes, and BDC. The ISA Summary (USEC, 2006a) also details the methods used by the applicant to identify hazards associated with the processes identified, contains descriptions of the accident sequences identified, evaluates the potential consequences for each accident, and identifies applicable IROFS for intermediate consequence events that are "Not Unlikely" and high consequence events that are "Not Unlikely" or "Unlikely." The safety function of each IROFS was described, along with the means by which the IROFS will be implemented.

The staff also reviewed selected portions of the ISA. The staff conducted four in-office reviews at the applicant's facility in Piketon, Ohio, for this part of the ISA review. The staff analyzed the applicant's proposed safety program, which includes the elements of process safety information, ISA, and management measures, to determine that the requirements of 10 CFR 70.62 are met. These in-office reviews also included vertical slice reviews of various accident sequences and horizontal reviews with respect to chemical safety, structural engineering, fire protection, radiation safety, criticality safety, human factors, and instrumentation and controls.[1] These reviews confirmed that applicant is adequately implementing the safety program and associated elements to achieve the performance requirements of 10 CFR 70.61. These reviews are described in Section A.3.2 of this SER.

In accordance with the guidance in Section 3.5.2.3 of NUREG-1520 (NRC, 2002), the vertical slice reviews examined how the ISA method was applied to a selected subset of facility processes, to obtain reasonable assurance that ISA methods would be effective in the other processes that the staff did not sample. The staff reviewed the applicant's combined Preliminary Hazard Analysis (PHA) and What if/Checklist (WI/CL) methodology and confirmed that it met the guidance of NUREG-1513 and generally acceptable industry practices (AIChE,1992). The PHA-WI/CL technique identifies and evaluates safety hazards in process plants and the technique may use either basic or detailed information concerning the design and operation of a process and may be used in various stages of the design process, as per NUREG-1513 (NRC, 2001). Implementation of this technique involves the use of an interdisciplinary team and a systematic approach to identifying hazards and operability problems that could lead to accident sequences of concern. The results of the PHA-WI/CL analysis are the team's findings, which include identification of the accident sequences and IROFS. As a result of the staff's review, additional accident sequences were added to the ISA and ISA Summary (USEC, 2006a) in the areas of fire safety, chemical safety, and nuclear criticality safety.

Accident sequences related to chemical safety, nuclear criticality safety, fire protection, and instrumentation and controls were selected for detailed vertical-slice reviews based on gas centrifuge uranium enrichment process knowledge and professional judgement. The vertical-slice reviews examined how the ISA methods were applied and examined appropriate safety information not included in the ISA Summary (USEC, 2006a). The vertical slice review included both high and intermediate consequence accident scenarios. The purpose of the reviews 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 likelihood to "highly unlikely," and each intermediate consequence event likelihood to "unlikely"). The results of the staff's vertical slice reviews of smart samples of accident sequences in major technical disciplines provide reasonable assurance that, if the methods described in the LA (USEC, 2006b) and ISA Summary (USEC, 2006a), as discussed above, are appropriately applied by the applicant, all accident sequences and related IROFS have been identified by the applicant or will be in its ongoing ISA process.

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[1]NRC's vertical slice reviews involved detailed reviews of individual accident sequences including review of the adequacy of IROFS. Horizontal reviews involved a broad consideration of the implementation of the ISA methodology to primarily determine whether credible accident sequences with intermediate or high consequences had been adequately identified.

3.3.1 Safety Program and ISA Commitments

The staff reviewed the applicant's proposed safety program commitments identified in Section 3.1 and Chapter 11 of the LA (USEC, 2006b) to determine that the three elements of process safety information, ISA, and management measures demonstrate compliance with the requirements of 10 CFR 70.62, and that records will be established and maintained for documenting each discovery that an IROFS or management measure has failed or degraded such that it cannot perform its intended safety function.

3.3.1.1 Process Safety Information

According to Section 3.1.1 of the LA (USEC, 2006b), the applicant will compile and maintain process information addressing:

- The hazards of materials used or produced in the process including information on chemical and physical properties (e.g., toxicity, acute exposure limits, reactivity, and chemical and thermal stability);

- The description of the technology of the process including block flow diagrams or simplified process flow diagrams, a brief outline of process chemistry, safe upper and lower limits for controlled parameters (e.g., temperature, pressure, flow, and concentration), and evaluation of the health and safety consequences of process deviations;

- Equipment used in the process, which includes general information on topics such as the materials of construction, piping and instrumentation diagrams, ventilation, design codes and standards employed, material and energy balances, IROFS, electrical classification, and relief system design; and

- The applicability of 29 CFR 1910.119, "Process Safety Management," and 40 CFR Part 68, "Risk Management Plan," to operation of the ACP to assure that chemicals not related to the licensed material are evaluated as necessary.

Program commitments require the compilation of up-to-date documentation of process safety information pertaining to the hazards of all materials used or produced in the process, the technology of the process, and the process equipment. The program commitments also require that process safety information be maintained up-to-date in accordance with the program elements described in Chapter 11 of the LA (USEC, 2006b). Chapter 11 of this SER discusses the staff review and evaluation of the proposed program elements needed to maintain the process safety information documentation.

As the design is finalized, more specific process safety information is expected to be developed. In Section 3.1.2 of the LA (USEC, 2006b), the applicant has committed to updating, as appropriate, its process safety information as the ACP design is finalized as follows:

> As the final design is developed for the ACP, the management system and design approach will require that the final designs be reviewed against the ISA to ensure the ISA accurately reflects the ACP design and operations, identifies the credible accident

sequences and appropriate assumptions, and credits the IROFS necessary to meet the performance requirements of 10 CFR 70.61.

Therefore, the staff concludes that if the program elements are followed, there is reasonable assurance that the applicant will compile and maintain process safety information up-to-date in accordance with the above mentioned commitment.

The applicant has committed to develop procedures and criteria for changing the ISA that meet the requirements of 10 CFR 70.72 in accordance with procedure development and implementation requirements contained in Section 11.1 of the LA (USEC, 2006b). Also, the ISA will be maintained and updated by personnel with the appropriate experience and expertise in engineering and process operations. The ISA Team for the various processes consists of individuals who are knowledgeable in the ISA method(s) and the operation, hazards, and safety design criteria of the particular process. Training and qualifications of individuals responsible for maintaining the ISA are described in Sections 2.1, 3.1.2, and 11.3 of the LA (USEC, 2006b) as well as ISA Summary Section 4.1 (USEC, 2006a).

Consistent with Section 3.4.3.1(1)(a) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains commitments to compile and maintain an up-to-date database of process safety information, and is, therefore, acceptable.

Consistent with Section 3.4.3.1(1)(b) of NUREG-1520 (NRC, 2002), the process safety element of the applicant's safety program includes procedures and criteria for changing the ISA along with a commitment to design and implement a facility change mechanism, and is, therefore, acceptable.

Consistent with Section 3.4.3.1(1)(c) of NUREG-1520 (NRC, 2002), the process safety element of the applicant's safety program contains a commitment to engage personnel with appropriate experience and expertise in engineering and process operations, and is, therefore, acceptable.

3.3.1.2 ISA Commitments

In Section 3.1 of the LA (USEC, 2006b), the applicant identifies ISA program commitments that were used to establish the ISA process. Those commitments include the performance of an ISA for each process that identifies the radiological hazards, chemical hazards that could increase radiological risk, chemical hazards from materials involved in processing licensed material, facility hazards that could increase radiological risk, potential accident sequences, consequences and likelihood of each accident sequence, and IROFS including the assumptions and conditions under which they support compliance with the performance requirements of 10 CFR 70.61. The staff's evaluation of the applicant's methods and criteria for implementing the ISA methodology is contained in Section 3.3.2 of this SER.

Adequate implementation of the Configuration Management Program described in Section 11.1 of the LA (USEC, 2006b) will ensure that the ISA and supporting documentation are accurately maintained and kept up-to-date. The ISA update process will account for any changes to the facility or its processes. The update process will also verify that initiating event frequencies and IROFS availability/reliability values assumed in the ISA remain valid. Management policies, organizational responsibilities, revision time frame, and procedures to perform and approve revisions to the ISA are outlined in LA Section 11.0 (USEC, 2006b). Proposed changes to the

facility will be evaluated using the ISA method(s). All IROFS will be maintained available and reliable, and unacceptable performance deficiencies will be addressed. The adequacy of existing IROFS and associated management measures will be promptly evaluated to determine if they are impacted by changes to the facility. Unacceptable performance deficiencies associated with IROFS will be addressed when identified through updates to the ISA. Written procedures will be maintained on-site in accordance with the requirements of Section 11.4 of the LA (USEC, 2006b). The staff's evaluation of the applicant's configuration management program is in Chapter 11 of this SER.

Consistent with Section 3.4.3.1(2)(a) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains a commitment to conduct an ISA of appropriate complexity for each process, and is, therefore, acceptable.

Consistent with Section 3.4.3.1(2)(b) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains a commitment to maintain the ISA and its supporting documentation so that it is accurate and up-to-date, and is, therefore, acceptable.

Consistent with Section 3.4.3.1(2)(c) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains a commitment to train personnel in the facility's ISA methods and/or use suitably qualified personnel to update and maintain the ISA and ISA Summary (USEC, 2006a), and is, therefore, acceptable.

Consistent with Section 3.4.3.1(2)(d) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains a commitment to evaluate proposed changes to the facility or its operations by means of the ISA method(s) and to designate new or additional IROFS and appropriate management measures as required, and is, therefore, acceptable.

Consistent with Section 3.4.3.1(2)(e) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains a commitment to address any IROFS' unacceptable performance deficiencies that are identified through updates to the ISA, and is, therefore, acceptable.

Consistent with Section 3.4.3.1(2)(f) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains a commitment to maintain written procedures on site, and is, therefore, acceptable.

Consistent with Section 3.4.3.1(2)(g) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains a commitment to establish all IROFS (if not already established) and to maintain them so that they are available and reliable when needed is evaluated in Section 3.3.1.3 of this SER, and is, therefore, acceptable.

3.3.1.3 Management Measures

In Section 7.1 of the ISA Summary (USEC, 2006a), the applicant describes management measures that comprise the principal mechanism by which the reliability and availability of each IROFS are ensured. General requirements applicable to each IROFS for configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigation, records management, and other quality assurance elements, are discussed. The incident investigation process will be responsible for the identification, reporting, and investigation of abnormal events. Where warranted, corrective actions will be carried out in accordance with the applicant's corrective action program. All records associated

with IROFS, and any item that may affect the function of IROFS, will be managed and controlled in a systematic manner to provide identifiable and retrievable documentation. The management measures are further detailed in Chapter 11 of the LA (USEC, 2006b), and evaluated in Chapter 11 of this SER. All plant components are assigned to one of their Quality Levels (QLs) as follows:

1. QL level 1 is assigned to a sole IROFS that prevents or mitigates a high consequence event;

2. QL level 2 is assigned to an IROFS that together with one or more other IROFS prevents or mitigates a high consequence event or by itself or with more IROFS prevents or mitigates an intermediate consequence event; and

3. QL level 3 is assigned to items other than QL-1 and QL-2; QL-3 items are controlled in accordance with standard commercial practices.

IROFS may only be assigned QL-1 or QL-2.

Consistent with Section 3.4.3.1(3)(a) of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) contains an acceptable commitment to establish management measures that comprise the principal mechanism by which the reliability and availability of each IROFS are ensured.

3.3.1.4 Safety Program and ISA Commitments Conclusion

Based on the above information, the staff concludes that the applicant meets the requirements of 10 CFR 70.62(a)(1) through (3) to establish and maintain a safety program that includes process safety information, ISA, and management measures.

3.3.2 ISA Methodology

The following sections describe the applicant's ISA methodology. This methodology is also described graphically as a flow diagram in the following Figures 3-1a through 3-1d.

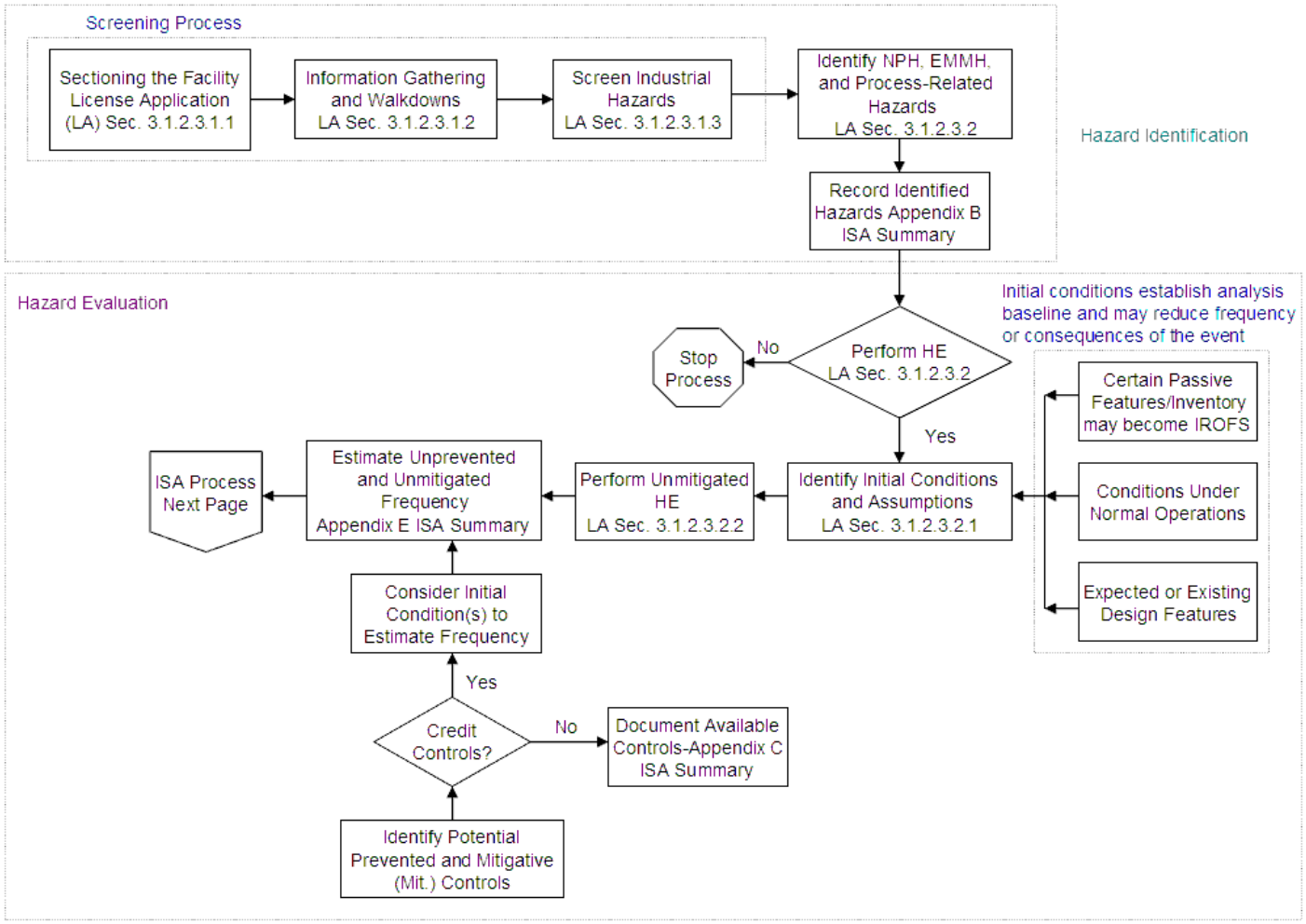

Figure 3-1a ISA for the ACP

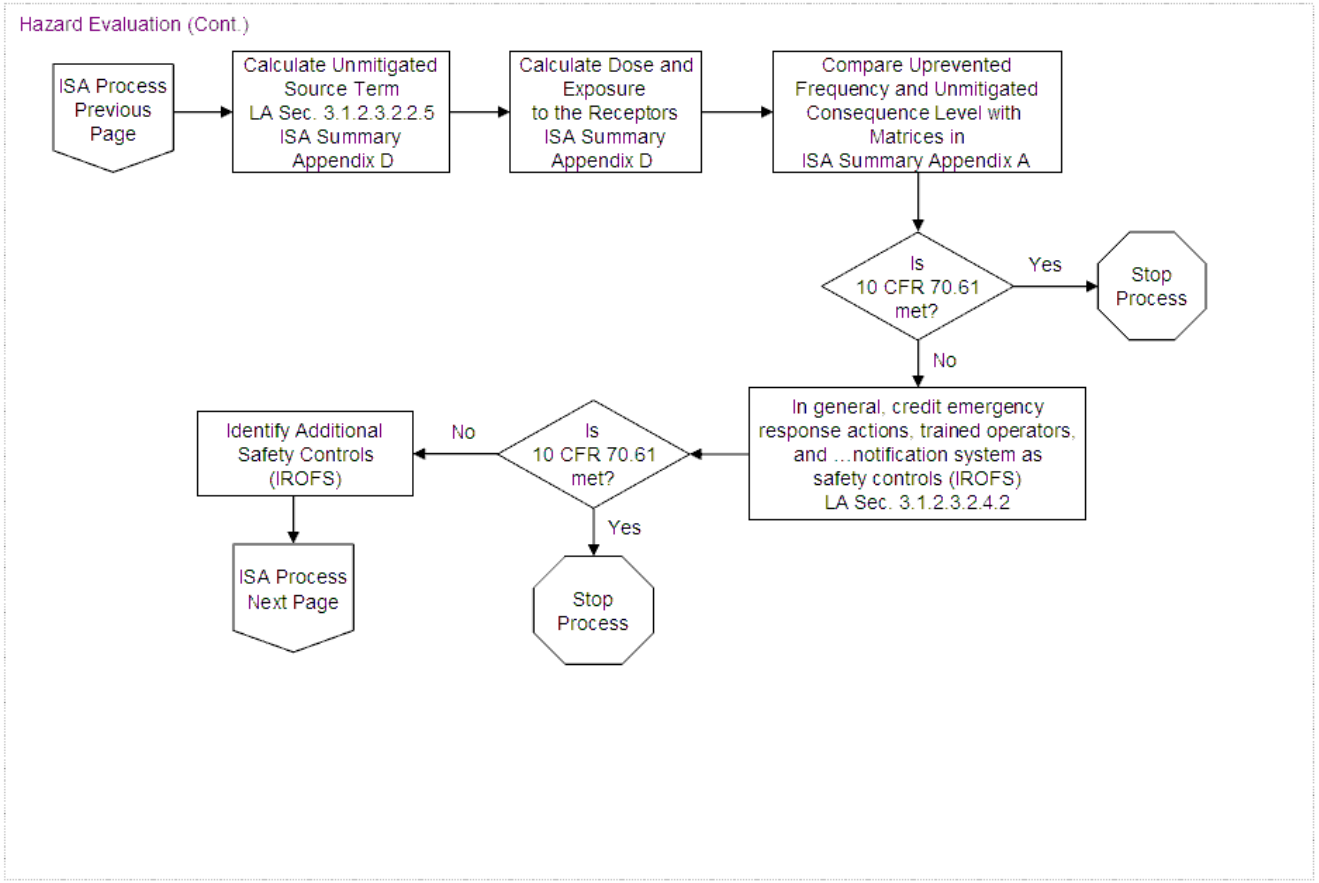

Figure 3-1b ISA for the ACP

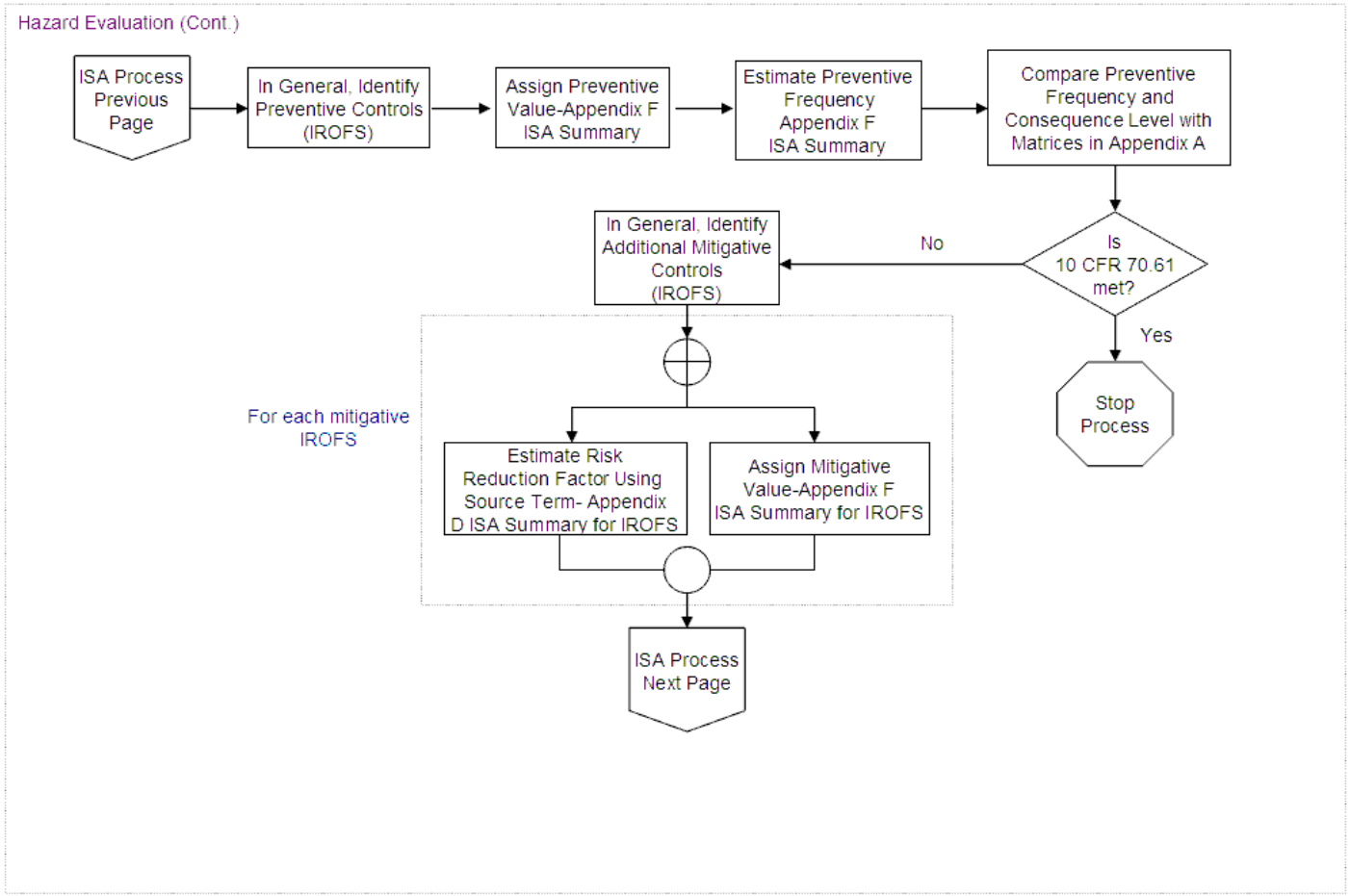

Figure 3-1c ISA for the ACP

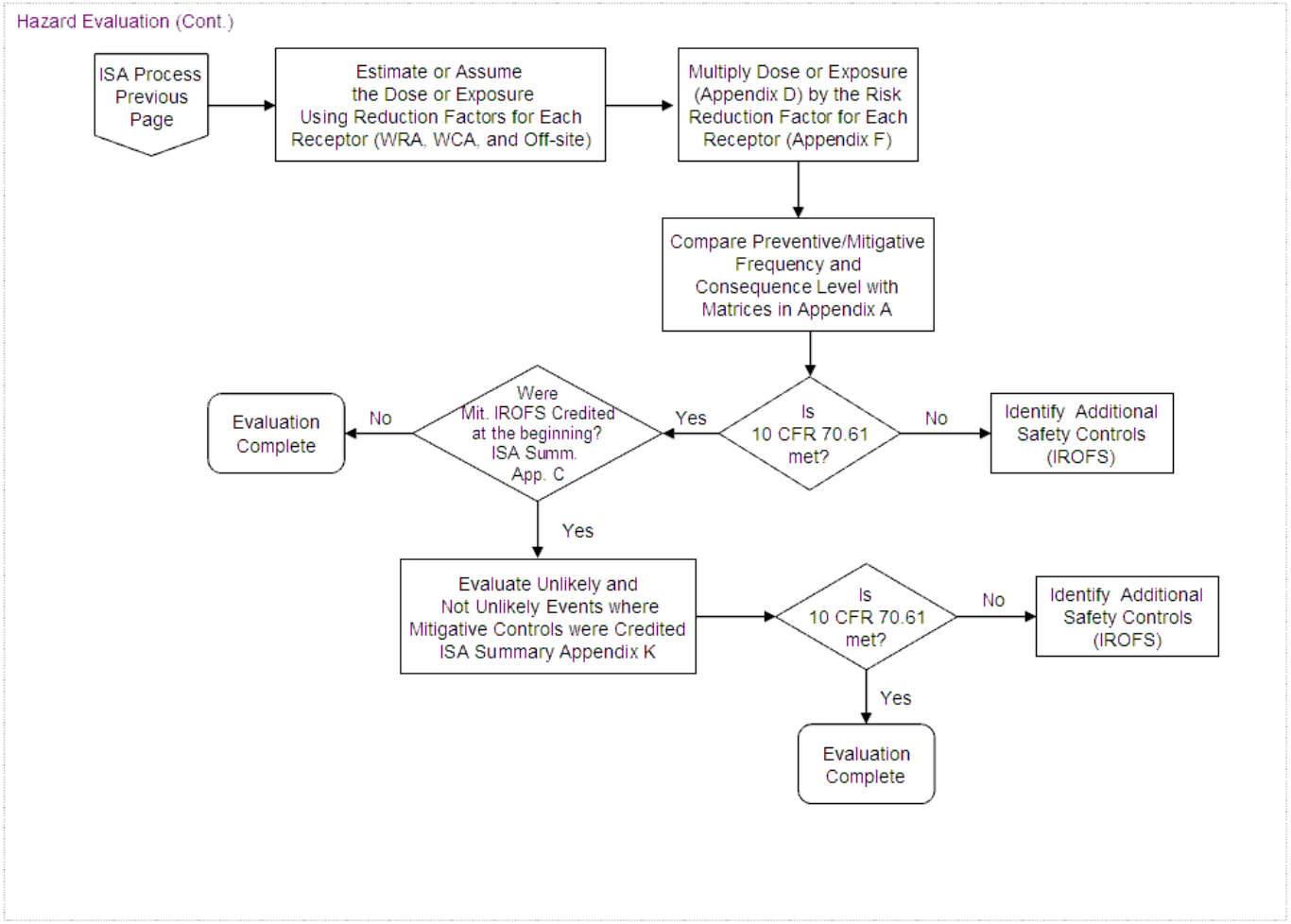

Figure 3-1d ISA for the ACP

3.3.2.1 Hazard Identification and Evaluation

The ISA analyzes the hazards associated with ACP operation, its associated direct support equipment and support systems, and the building and facilities where it is located. Hazards associated with sabotage or chemical hazards that do not result from the processing of licensed nuclear material or do not have the potential for adversely affecting radiological safety are not addressed.

The applicant used the guidelines presented in Appendix A of NUREG-1513 (NRC, 2001) as a basis for selecting the Hazard Evaluation Method, using the methodology in the flowchart presented as Figure A.1 of NUREG-1513 (NRC, 2001). Using these guidelines, the ISA team selected a hybrid method that incorporated elements of the PHA and the WI/CL method.

The hazard analysis for the ISA is developed using two primary activities: (i) hazard identification and (ii) hazard evaluation. These activities result in the estimates of consequences and frequencies such that scenarios can be identified that have a potential risk to the public, workers, and the environment that exceeds the 10 CFR 70.61 performance requirements. The hazard identification process is shown in the top boxed-in set of ISA process steps in Figure 3-1a. The hazard evaluation process is comprised of the rest of the process steps shown in Figures 3-1a - 3-1d.

Hazard identification is a comprehensive and systematic process by which all known hazards associated with the facility and processes are identified, recorded, and screened by the ISA team. To perform the hazard identification, the facility was partitioned into the following sections:

- Cylinder Storage Areas;
- Feed Area of Feed and Customer Services Building;
- Interconnecting Process Piping;
- Process Buildings includes Process Support Building;
- Product and Tails Withdrawal Building;
- Recycle/Assembly Building/Centrifuge Training and Testing Facility/Interplant Transfer Corridor;
- Sampling and Transfer Area of Feed and Customer Services Building;
- Transportation Activity;
- Feed and Product Shipping and Receiving Building; and
- Criticality Events.

For each of these sections, information gathering, including paper walkdowns, physical walkdowns, and operator interviews, were used to create a list of all expected radiological and chemical hazards. The hazard identification and evaluation process considered the applicable ACP activities including startup, normal operation, shutdown, and maintenance activities as well as concurrent construction activities. As part of the hazard identification process, chemical and standard industrial processes were screened. The applicant's screening methodology eliminated hazards that would not present a significant threat to health or safety or would be sufficiently covered by Occupational Safety and Health Administration (OSHA) regulations or applicable national consensus standards. In addition to the process hazards, natural phenomena hazards (NPH) (e.g., earthquakes, tornadoes, straight winds) and external man-made hazards (EMMH) (e.g., aircraft and vehicular impact) are also identified and listed in

Appendix B of the ISA Summary (USEC, 2006a)

The purpose of the hazard evaluation is to ensure a comprehensive assessment of facility hazards and to focus attention on those events that pose the greatest risk to the public and on-site workers. The scope of the hazard evaluation includes:

- Incorporation of identified aspects of facility processes and operations;
- Incorporation of NPH and EMMH and nuclear criticality (where applicable);
- Consideration of the entire spectrum of possible events for a given hazard in terms of both frequency and consequence levels; and
- Hazards addressed by other programs and regulations (e.g. OSHA, Department of Energy (DOE), Environmental Protection Agency (EPA)) if loss of control of the hazard could result in a release of radiological material or hazardous chemicals.

The hazard evaluation process is divided into the following steps for the purpose of developing accident sequences:

- Identification of initial conditions and assumptions;
- Unmitigated hazard evaluation; and
- Mitigated hazard evaluation.

ICs are assumptions that are used to establish an analysis reference baseline during an evolving design or clarify a point of analysis that might otherwise be unstated. ICs are established from operating conditions and limitations under which the ACP is anticipated to operate. ICs typically delineate specific conditions that are part of normal facility operations or delineate specific features of the facility that are unlikely to change and are used in establishing the unmitigated frequencies, consequences, or types of events. To preserve the integrity of ICs, they are credited and treated as IROFS.

In general, ICs represent assumptions made in the analysis for probability or consequences or they are specific design features credited in the probability analyses. Three examples are: 1) the header isolation features which serve to limit the material at risk as assumed in the consequence analysis; 2) the combustible materials control program which serves to limit the presence of material that could fuel facility fires; and 3) the structural seismic specifications which serve to establish minimum structural requirements to reduce the frequency of serious damage caused by earthquakes.

ICs that are associated with a specific or a limited number of events are identified in the event description of those events in bold type font followed by IROFS numbers. Initial conditions that apply to many events, such as cylinder integrity specifications, are not repeated in the event description of each event except for criticality events where all applicable ICs are identified.

The unmitigated hazard evaluation postulates events that could occur within, or otherwise impact the facility, and assigns event frequencies and event consequences without regard to preventative or mitigative design features or programs. The applicant categorized events according to the nature of the postulated release mechanism. Categories of events considered include:

- Fire;
- Explosion;
- Loss of containment/confinement;
- Direct radiological/chemical exposure;
- Nuclear criticality;
- External hazards; and
- Natural phenomena.

Information related to the unmitigated hazard evaluation is collected and organized in hazard evaluation tables. These tables address the non-screened hazards associated with the systems and areas identified during the hazard identification process. These tables are located in Appendix C of the ISA Summary (USEC, 2006a) and contain:

- Event number and category;
- Event description (including location, release mechanism, material at risk, initial conditions, and hazard source);
- Causes;
- Unprevented event frequency level;
- Unmitigated consequence level (categorized as low, intermediate, or high); and
- Unprevented/unmitigated risk bin (categorized as above or below low threshold limits).

In addition, any preventive or mitigative controls that may be available within the facility are listed, however, no credit is taken for these controls during the unmitigated hazard analysis unless the control is listed as an initial condition. The initial conditions are taken into account for the estimation of unmitigated frequency. Inputs and assumptions used in the unmitigated frequency analysis are contained in Appendix E of the ISA Summary (USEC, 2006a). Sources of event frequency are referenced and include generic initiator database information and failure data from other sites such as the Savannah River Site (Coutts et al, 2000) and engineering calculations, etc. The “unprevented” frequency level evaluations are summarized in Table A-4 of the ISA Summary (USEC, 2006a), as “Not Unlikely,” “Unlikely,” and “Highly Unlikely.” The ISA team assigns a frequency level based on the sources of frequency-related information, the methods used to evaluate that information, and the uncertainties associated with the evaluation process. The qualitative frequency level is then fed back to the hazard evaluation tables in Appendix C of the ISA Summary (USEC, 2006a). The determination of the unprevented frequency is shown as the last step in the ISA process as shown in Figure 3-1a.

Unmitigated consequences are generally conservative calculations of dose or exposure at specified receptor locations with the exception of the criticality events which are assumed to result in high consequences. Unmitigated consequences consider any mitigative ICs and are based on (1) simple source term calculations, (2) existing safety documentation, and (3) qualitative assessment. The methods used are consistent with those described in NUREG/CR-6410 (NRC, 1998). Inputs and assumptions used in the unmitigated consequence analysis are provided in Appendix D of the ISA Summary (USEC, 2006a). The results of the unmitigated consequence analysis are compared with the criteria established in Tables A-5 and A-6 in Appendix A of the ISA Summary (USEC, 2006a) and the results are fed back into the hazard evaluation tables. This part of the ISA process is shown on the upper part of the flow chart in Figure 3-1b. If the event sequence satisfies the 10 CFR 70.61 performance criteria, the analysis for that event sequence stops, the sequence is recorded in Appendix C of the ISA Summary (USEC, 2006a), and no further analysis is required.

If event risk to the public or workers exceeds the 10 CFR 70.61 performance requirements, a more detailed analysis may be conducted to refine the event frequency and consequences for the event of concern. This additional analysis involves the application of controls either mitigative, preventive, or both to the sequence. For sequences that involved radiological and/or chemical releases (not nuclear criticality), the first controls to be credited were the event response controls, which include emergency response actions; alarm, notification and protective actions; and trained operator actions. These controls are mitigative and will reduce the consequences to the worker two orders of magnitude (10^{-2}), and one order of magnitude (10^{-1}) for the public. The likelihood and consequences are again compared with the 10 CFR 70.61 performance criteria and if these criteria are satisfied the analysis ends for that sequence. Although these mitigative controls are often used first because of their generic nature and ease of application, the ISA methodology generally applies preventive controls next before additional mitigative controls as discussed in Section 3.1.2.3.2.4.2 of the LA (USEC, 2006b).

If the 10 CFR 70.61 performance criteria are not satisfied, the process progresses as shown at the start of the process as shown in Figure 3-1c where available preventive controls are credited to meet the performance requirements. These controls are selected in an order of preference. If available, engineered or designed controls will be selected before administrative controls, to use the inherent reliability advantage of designed systems or components over that of required human action compliance. Defense-in-depth concepts using non-credited controls were also incorporated into the control strategy for a postulated event whenever possible according to Section 3.1.2.3.2.4 of the LA (USEC, 2006b). These defense-in-depth controls are generally controls that were listed as available when the sequences were formulated but not explicitly required to meet the performance requirements.

The effects of the various preventive controls on an event or accident sequence are termed preventive values and are addressed in Appendix F of the ISA Summary (USEC, 2006a). Typical preventive values are three orders of magnitude (10^{-3}) for a passive engineered control, two orders of magnitude (10^{-2}) for an active engineered control, and one order of magnitude (10^{-1}) for an administrative control. These values are consistent with values provided in Table A-10 of NUREG-1520 (NRC, 2002). If the performance requirements regarding likelihood could not be met using a combination of preventive controls, then additional mitigative controls to lower the consequences to receptors were applied. The effects of the various mitigative controls on an accident sequence are termed mitigation values and are also addressed in Appendix F of the ISA Summary (USEC, 2006a). The values of the various mitigative controls are derived in Appendix D of the ISA Summary (USEC, 2006a). Together, the values of preventive controls and mitigative controls are known as risk reduction factors.

The final steps in the ISA analysis are shown in Figure 3-1d. This step of the analysis evaluates the likelihood of failure of mitigative IROFS for events that are not prevented or made to be "highly unlikely." Events for which mitigative controls were credited were evaluated to examine the residual risk associated with the postulated failure upon demand of each mitigative IROFS. The approach used in the evaluation developed a series of sub-events designed to demonstrate that the risk of the event after failure of one or more mitigative controls is still within the 10 CFR 70.61 performance requirements. This evaluation is described and tabulated in Appendix K of the ISA Summary (USEC, 2006a).

Consistent with Sections 3.4.3.1(2)(a), 3.4.3.2(3), and 3.4.3.2(5)(a) of NUREG-1520 (NRC,

2002), the applicant's hazard identification method: (i) provided a list of materials and conditions that could result in hazardous situations; and (ii) determined potential interactions between materials or conditions that could result in hazardous situations, and is, therefore, acceptable.

Consistent with Sections 3.4.3.2(3) and 3.4.3.2(5)(b) of NUREG-1520 (NRC, 2002), the applicant's process hazard analysis method involves selecting one of the methods described in NUREG-1513 (NRC, 2001) and applying it in accordance with the selection criteria established in that document, and is, therefore, acceptable.

Consistent with Sections 3.4.3.2(4) and 3.4.3.2(5)(c) of NUREG-1520 (NRC, 2002), the applicant's consequence evaluation methods are: (i) in accordance with the approaches in NUREG/CR-6410 (evaluated in Section 6.3.3 of this SER); and (ii) use generic assumptions and data in a reasonably conservative manner for the types of accidents analyzed, and are, therefore, acceptable.

3.3.2.2 Definition of Receptors for Consequence Evaluations

The receptors for chemical and radiological consequences and their locations are described as follows:

> Off-Site (public) - outside the site boundary or controlled area. Off-site doses or chemical exposures are conservatively estimated for the public at a distance from the point of release to the nearest site boundary as tabulated in Subsection 4.4.2.2.5 of the ISA Summary (USEC, 2006a).
>
> WCA - workers in the controlled area are workers typically outside the restricted area but within the controlled area of the site boundary. For evaluation purposes, these workers are located outside the last possible barrier from the hazard and at the worst possible location. Doses or chemical exposures are estimated for the WCA receptor at a distance of 100 meters. Typically, this would represent a point near to the exterior walls of the analyzed facility, but far enough outside that releases could have the potential to reach ground level.
>
> WRA - workers in the restricted area are workers inside the facility. This category of receptors includes those in the immediate area of the hazard, and those workers in the same room or building who may not be aware of the hazardous condition but would quickly become aware of the hazardous condition and take appropriate action. Doses or chemical exposures are estimated qualitatively, but in all cases it is assumed that the WRA receives a dose at least as significant as the dose received by the WCA.

The controlled area is defined in Section 2.4.1 of the ISA Summary (USEC, 2006a) as the area outside of the restricted area but inside the DOE reservation boundary. The restricted areas, as defined in Section 2.4.2 of the ISA Summary (USEC, 2006a), are those locations where licensed material is present in quantities of concern regarding to the health and safety of the workers.

The quantitative dose limits the applicant assumes, to determine consequences to the public, workers, and the environment are presented in Section 3.3.2.6 of this SER.

3.3.2.3 Likelihood Evaluation Method

Likelihood Determination for Accident Sequences

The likelihood evaluation method is discussed in Section 3.1.2.3.2.2.4.1 of the LA (USEC, 2006b). The definitions of "not unlikely," "unlikely," and "highly unlikely" are provided in this Section as well as Table A-4 of the ISA Summary (USEC, 2006a). "Not unlikely" is defined as an event with a probability of occurrence of more than 10^{-4} per event per year. "Unlikely" is defined as an event with a probability of occurrence of between 10^{-4} and 10^{-5} per event per year. "Highly unlikely" is defined as an event with a probability of occurrence of less than 10^{-5} per event per year.

Not-Credible Determination

In Section 3.1.2.3.2.2.4.1 of the LA (USEC, 2006b), the applicant incorporates the definitions of "not credible" from NUREG-1520 (NRC, 2002), as follows:

A. An external event for which the frequency of occurrence can conservatively be estimated as less than once in a million years ($<10^{-6}$);

B. A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive (In determining that there is no reason for such actions, a wide range of possible motives, short of intent to cause harm, must be considered. Necessarily, no such sequence of events can ever have actually happened in any fuel cycle facility for processes similar to ACP processes); or

C. Process deviations for which there is a convincing argument, given physical laws that they are not possible, or are unquestioningly extremely unlikely (The validity of the argument must not depend on any feature of the design or materials controlled by the facility's system of IROFS or management measures).

The staff concludes that the above definitions of "not credible" are suitable for use in determining the likelihood performance requirements of 10 CFR 70.61.

Likelihood Evaluation Method

The proposed likelihood definitions are consistent with NUREG-1520, Chapter 3, Appendix A, Table A-6 (NRC, 2002). Implementation of the evaluation method is discussed in Section 3.1.2.3.2.5 of the LA (USEC, 2006b). Implementation is consistent with the likelihood evaluation methodology described in Appendix A of Chapter 3 of NUREG-1520 (NRC, 2002). The tables in Appendix C of the ISA Summary (USEC, 2006a) show how IROFS act to prevent or mitigate the consequences of an accident sequence. The likelihood of failure was evaluated for each IROFS as described in Section 3.3.2.1 of this SER.

Consistent with Section 3.4.3.2(5)(d) of NUREG-1520 (NRC, 2002), the applicant's likelihood evaluation method was performed in accordance with the guidance provided in Appendix A of Chapter 3 of NUREG-1520 (NRC, 2002), and is, therefore, acceptable.

Consistent with Section 3.4.3.2(9) of NUREG-1520 (NRC, 2002), the applicant's definitions of "Unlikely," "Highly Unlikely," and "Credible" are based on NRC's risk performance guidelines provided in the NUREG-1520 (NRC, 2002), and are, therefore, acceptable.

3.3.2.4 Chemical Consequences

This section evaluates the proposed chemical quantitative risk levels to protect the workers and the public. Radiological consequence limits are specified in 10 CFR 70.61(b) and (c). Chemical consequence limits, however, are not specified in the regulation, and in accordance with 10 CFR 70.65(b)(7) are proposed by the applicant as a part of the ISA Summary (USEC, 2006a) and evaluated by the staff. Accordingly, the applicant proposed chemical consequence levels for hydrogen fluoride and soluble uranium.

The proposed chemical consequences were binned as high, intermediate, or low based on Emergency Response Planning Guideline (ERPG) levels. The ERPGs are developed by the American Industrial Hygiene Association as guidance for emergency response personnel. These emergency response limits are defined in NUREG-1391 (NRC, 1991) as follows:

ERPG -1 "The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing other than mild, transient adverse, effects or without perceiving a clearly defined objectionable odor."

ERPG -2 "The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair the individual's ability to take protective action."

ERPG -3 "The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects."

The comparison of the ERPGs with the calculated airborne concentrations allows the applicant to assign a high, intermediate, or low chemical consequence level as shown in Table 3-1. In the case of exposures to soluble forms of uranium (i.e., UF_6 and uranyl fluoride (UO_2F_2)), the applicant, in its LA (USEC, 2006b), uses intakes of uranium expressed in milligrams (mg) of material.

Table 3-1
Consequence Severity as Related to ERPG Values

Consequence Level	**Off Site**	**Worker Inside the Controlled Area[a]**	**Worker Inside the Controlled Area[a]**
High	Concentration ≥ ERPG-2; or Intake ≥ 30mg of soluble uranium or which could lead to irreversible or serious long-lasting health effects.	Concentration ≥ ERPG-3; or intake ≥ 40mg of soluble uranium or which could endanger the worker's life.	Concentration ≥ ERPG-3; or intake ≥ 40mg of soluble uranium or which could endanger the worker's life.
Intermediate	ERPG-1 ≤ Concentration < ERPG-2; or intake ≥ 10mg of soluble uranium or which could cause mild transient health effects.	ERPG-2 ≤ Concentration < ERPG-3; or intake ≥ 30mg of soluble uranium or which could lead to irreversible or serious long-lasting health effects.	ERPG-2 ≤ Concentration < ERPG-3; or intake ≥ 30mg of soluble uranium or which could lead to irreversible or serious long-lasting health effects.
Low	Concentration < ERPG-1	Concentration < ERPG-2; or which could cause mild or transient health effects.	Concentration < ERPG-2; or which could cause mild or transient health effects.

[a] The worker inside the controlled area is outside the restricted area.

Specific chemical consequence levels proposed by the applicant are shown in Table 3-2.

Table 3-2
Chemical Consequence Levels Proposed by the Applicant

Chemical	**ERPG-1**	**EPRG-2**	**ERPG-3**
Soluble Uranium	10 mg	30 mg	40 mg
HF (1 hour)	1.5 mg/m^3	16.4 mg/m^3	41 mg/m^3
HF (10 minutes)	1.5 mg/m^3	41 mg/m^3	139 mg/m^3

Consistent with Section 3.4.3.2(7) of NUREG-1520 (NRC, 2002), the applicant's quantitative standards for chemical consequences meet the following criteria, and are, therefore, acceptable:

- There are unambiguous quantitative standards for each of the applicable hazardous chemicals that meet the criteria of 10 CFR 70.65(b)(7) on site;
- The quantitative standard of 10 CFR 70.61(b)(4)(i) addresses exposures that could endanger the life of a worker;
- The quantitative standards for 10 CFR 70.61(b)(4)(ii) and 10 CFR 70.61(c)(4)(i) will correctly categorize all exposures that could lead to health effects on individuals that could be long lasting; and

- The quantitative standard for 10 CFR 70.61(c)(4)(ii) will correctly categorize all exposures that could cause individual mild transient health effects.

3.3.2.5 Radiological Consequences

Radiological consequences are specified in 10 CFR 70.61(b) and (c) and are presented in Table 3-3.

Table 3-3
Radiological Consequence Categories as Specified in 10 CFR Part 70

Consequence level	Off Site	Workers Inside the Controlled Area	Worker Inside the Restricted Area
High	≥25 rem (0.25 Sv)	≥100 rem (1 Sv)	≥100 rem (1 Sv)
Intermediate	≥5 rem (0.05 Sv)	≥25 rem (0.25 Sv)	≥25 rem (0.25 Sv)
Low	<5 rem (0.05 SV)	<25 rem (0.25 Sv)	< 25 rem (0.25 Sv)

3.3.2.6 Environmental Consequences

The applicant addressed environmental consequences in Appendix I of the ISA Summary (USEC, 2006a). 10 CFR 70.61(c) requires that the risk for credible events be limited so that concentrations outside the restricted area for a 24-hour averaged release do not exceed 5,000 times the levels indicated in Table 2 of Appendix B in 10 CFR Part 20. Table 2 of Appendix B identifies the concentration values for radioactive isotopes that a person would have to be exposed to for a 1-year period to result in a dose of 0.05 rem. The concentration of uranium isotopes in UF_6 or UO_2F_2 that corresponds to the performance requirement is $1.5X10^{-8}$ microcuries per milliliter. The release consequence to the environment is considered only as an intermediate consequence. The staff reviewed the applicant's methodology and assumptions for determining compliance with the environmental consequences requirement and finds them to be acceptable. The staff also made confirmatory calculations for selected scenarios as discussed in Appendix B of this SER.

3.3.2.7 ISA Methodology Conclusion

Based on the above information, the staff concludes that the applicant has used a methodology adequate to identify hazards related to this type of facility and credible events that could exceed the performance requirements of 10 CFR 70.61. The applicant has also established appropriate definitions of likelihood and applied those definitions in an acceptable manner to demonstrate that intermediate consequence events are unlikely and high consequence events are highly unlikely.

3.3.3 Compliance with the BDC and Defense-In-Depth Requirements

The ACP is a new facility and, therefore, must comply with 10 CFR 70.64(a) and (b) which include BDC and Defense-In-Depth practices. A review of compliance with these regulatory requirements is included below.

3.3.3.1 BDC

10 CFR 70.64(a) requires each applicant for a new facility or for a new process at an existing facility to address 10 BDC. Licensees are required to maintain the application of these criteria unless the analysis performed pursuant to 10 CFR 70.62(c) demonstrates that a given item is not relied upon for safety or does not require adherence to the specified criteria. The following is a list of the required BDC and how the applicant has met these criteria:

Quality Standards and Records [BDC (1)]. The design must be developed and implemented in accordance with management measures, to provide adequate assurance that IROFS will be available and reliable to perform their function when needed. Appropriate records of these items must be maintained by or under the control of the licensee throughout the life of the facility. Management measures and quality elements such as design control, procurement control, testing, corrective action program, record keeping, audits, and other elements are discussed in Chapter 11.0 of this SER. Based on that evaluation, the staff determined that the applicant has appropriately addressed BDC (1).

Natural Phenomena Hazards [BDC (2)]. The design must provide for adequate protection against natural phenomena with consideration of the most severe documented historical events for the site. Natural phenomena that could affect the site are discussed in Chapter 1.0 and Section A.3.1.1 of this SER. Based on that evaluation, the staff determined that the applicant has appropriately addressed BDC (2).

Fire Protection [BDC (3)]. The design must provide for adequate protection against fires and explosions. Fires and explosions are discussed in Section 7.3 and Appendix D of this SER. Based on that evaluation, the staff determined that the applicant has appropriately addressed BDC (3).

Environmental and Dynamic Effects [BDC (4)]. The design must provide for adequate protection from environmental conditions and dynamic effects associated with normal operations, maintenance, testing, and postulated accidents that could lead to loss of safety functions. Environmental and dynamic effects as they affect electrical equipment are discussed in Section E.3.4 of this SER. They are also addressed in Section A.3.1.5 of this SER in regard to the applicant's IROFS Boundary Determination Plan (USEC, 2005). Based on that evaluation, the staff determined that the applicant has appropriately addressed BDC (4).

Chemical Protection [BDC (5)]. The design must provide for adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material. Chemical protection is evaluated in Section 6.3.6 of this SER. Based on that evaluation, the staff determined that the applicant has appropriately addressed BDC (5).

Emergency Capability [BDC (6)]. The design must provide for emergency capability to maintain control of:

- Licensed material and hazardous chemicals produced from licensed material;
- Evacuation of on-site personnel; and
- On-site emergency facilities and services that facilitate the use of available off-site services.

Emergency planning is addressed in Chapter 8 of this SER. Based on that evaluation the staff has determined that the applicant has appropriately addressed BDC (6).

Utility Services [BDC (7)]. The design must provide for continued operation of essential utility services. Utility services are discussed in Section E.3.3 of this SER. Based on that evaluation, the staff has determined that the applicant has appropriately addressed BDC (7).

Inspection, Testing, and Maintenance [BDC (8)]. The design of IROFS must provide for adequate inspection, testing, and maintenance to ensure their availability and reliability to perform their function when needed. Inspection, testing, and maintenance are discussed in Chapter 11.0 of this SER. Based on that evaluation, the staff has determined that the applicant has appropriately addressed BDC (8).

Criticality Control [BDC (9)]. The design must provide for criticality control including adherence to the double contingency principle. Criticality control in regard to the BDC is addressed in Section 5.3.10 of this SER. Based on that evaluation, the staff has determined that the applicant has appropriately addressed BDC (9).

Instrumentation and Controls [BDC (10)]. The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of IROFS. Instrumentation and controls are discussed in Section E.3.2.1 of this SER. Based on that evaluation, the staff has determined that the applicant has appropriately addressed BDC(10).

The staff conducted an on-site review of both the classified and non-classified portions of the ISA and concludes that the applicant has appropriately addressed the BDC for new facilities, as required for 10 CFR 70.64(a).

3.3.3.2 Defense-In-Depth

The facility design and operations are based on defense-in-depth practices. Controls are selected with a preference for engineered controls over administrative controls when available (Section 3.1.2.3.2.4.2 of the LA (USEC, 2006b)). Controls that are not credited to demonstrate compliance with 10 CFR 70.61 were incorporated into the control strategy for a postulated event whenever possible. These non-credited controls are listed in regular type in Appendix C of the ISA Summary (USEC, 2006a) under the lists of controls whereas IROFS are listed in bold type. In many cases, these non-credited controls are expected to reduce challenges to IROFS in the case of an initiating event. The staff concludes that the applicant has appropriately addressed the defense-in-depth practices for new facilities as required by 10 CFR 70.64(b).

3.3.3.3 Conclusion for BDC and Defense-in-Depth

Based on the above, and consistent with Section 3.4.3.2(4)(d) of NUREG-1520 (NRC, 2002), the staff concludes that the applicant has provided sufficient information in the ISA Summary (USEC, 2006a) to demonstrate that the BDC were applied in an acceptable manner to the design of the gas centrifuge uranium enrichment facility and the facility and system design and facility layout will be based on defense-in-depth practices.

3.4 EVALUATION

The staff finds the applicant's maintenance of process safety information is in accordance with the guidance of NUREG-1520 (NRC, 2002).

The staff finds that the applicant's commitment to conduct and maintain an ISA is in accordance with the requirements of 10 CFR 70.62(c)(1) and the guidance in NUREG-1520 (NRC, 2002).

The staff considers the ISA methodology to be complete by its use of the appropriate accident identification methodology from NUREG-1513 (NRC, 2001). The staff considers the consequence determinations to be acceptable and in accordance with the guidance in NUREG/CR-6410 (NRC, 1998). The staff has also evaluated the consequence determination methodology and results are also evaluated by the staff in Appendix B of this SER. The staff considers the likelihoods to have been derived using acceptable methods and to comply with acceptable definitions of "not unlikely," "unlikely," and "highly unlikely" as evaluated in Appendix A of this SER. The staff concludes that these descriptions conform with the guidance provided in NUREG-1520 (NRC, 2002) and meet the requirements of 10 CFR 70.65(b)(4).

Area boundaries, including the controlled area boundary and the locations of restricted areas, are adequately described for the purpose of determining consequences and meet the requirements of 10 CFR 70.65(b)(2).

The likelihood evaluation, including definitions of "not credible" conforms with the guidance provided in NUREG-1520 (NRC, 2002) and is acceptable.

The determination of chemical consequences conforms with the guidance provided in NUREG-1520 (NRC, 2002) and is acceptable.

The staff concludes that the applicant has appropriately addressed the BDC for new facilities as required by 10 CFR 70.64(a) and has appropriately addressed the defense-in-depth practices for new facilities, as required by 10 CFR 70.64(b).

3.5 REFERENCES

(AIChE,1992) American Institute of Chemical Engineers (AIChE). "Guidelines for Hazard Evaluation Procedures, Second Edition with Worked Examples," 1992.

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

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

(NRC, 2001) U.S. Nuclear Regulatory Commission (NRC). NUREG-1513 "Integrated Safety Analysis Guidance Document," May 2001.

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

(USEC, 2005) USEC Inc. (USEC). AET 05-0081, “Submittal of Additional Information Related to the Integrated Safety Analysis for the American Centrifuge Plant,” October 28, 2005.

(USEC, 2006a) USEC Inc. (USEC). “Integrated Safety Analysis Summary for the American Centrifuge Plant in Piketon, Ohio,” Revision 14, August 2006.

(USEC, 2006b) USEC Inc. (USEC). “License Application for the American Centrifuge Plant in Piketon, Ohio,” Revision 18, September 2006.

4.0 RADIATION PROTECTION

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's radiation protection (RP) program is to evaluate whether the application provides adequate information to protect the radiological health and safety of workers and is in compliance with the associated regulatory requirements in 10 CFR Part 70. Public and environmental protection is discussed in Chapter 9 of this Safety Evaluation Report (SER).

4.1 REGULATORY REQUIREMENTS

4.1.1 RP Program Implementation

Regulations applicable to establishment of an RP program are presented in 10 CFR Part 20, Subpart B, "Radiation Protection Programs."

4.1.2 As Low As is Reasonably Achievable (ALARA) Program

Regulations applicable to the ALARA program are presented in 10 CFR 20.1101, "Radiation Protection Programs."

4.1.3 Organization and Personnel Qualifications

Regulations applicable to the organization and qualifications of the radiological protection staff are presented in 10 CFR 30.33, 10 CFR 40.32, and 10 CFR 70.22, "Contents of applications."

4.1.4 Written Procedures

The regulations applicable to RP procedures and Radiation Work Permits (RWPs) are presented in 10 CFR 30.33, 10 CFR 40.32, and 10 CFR 70.22, "Contents of applications."

4.1.5 Training

The following regulations apply to the radiation safety training program:

- 10 CFR 19.12 "Instructions to workers"; and
- 10 CFR 20.2110 "Form of records."

4.1.6 Ventilation and Respiratory Protection Programs

Regulations applicable to the ventilation and respiratory protection programs are presented in 10 CFR Part 20, Subpart H, "Respiratory protection and controls to restrict internal exposure in restricted areas."

4.1.7 Radiation Survey and Monitoring Programs

The following NRC regulations in 10 CFR Part 20 are applicable to radiation survey and monitoring programs:

- Subpart C "Occupational Dose Limits";
- Subpart F "Surveys and Monitoring";
- Subpart L "Records"; and
- Subpart M "Reports."

4.1.8 Additional Program Requirements

The following regulations are applicable to the additional program requirements:

- Subpart L "Records" of 10 CFR Part 20;
- Subpart M "Reports" of 10 CFR Part 20;
- 10 CFR 70.61 "Performance requirements"; and
- 10 CFR 70.74 "Additional reporting requirements."

4.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the RP section of the license application (LA) (USEC, 2006b) is contained in Chapter 4 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). Chapter 4 of NUREG-1520 (NRC, 2002) is applicable in its entirety. The acceptance criteria applicable to this review are contained 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 NUREG-1520 (NRC, 2002).

4.3 STAFF REVIEW AND ANALYSIS

In Chapter 4.0 of the LA (USEC, 2006b) for the American Centrifuge Plant (ACP), the applicant describes the ACP RP program. The applicant states in its application that the principles of the program will be in accordance with the ALARA principle, and that no individual will receive a radiation dose in excess of any regulatory limit. The responsibility for establishing these principles rests with the Vice President, American Centrifuge Plant. The Director, American Centrifuge Plant, has the overall responsibility and authority for the ALARA Program and the RP Manager (RPM) will be responsible for establishing and implementing the ALARA Program. The program is based on the American Centrifuge Lead Cascade Facility RP program, which was based on the Portsmouth Gaseous Diffusion Plant RP program (NRC, 1999). As part of licensing the Lead Cascade Facility under 10 CFR Part 70, NRC has reviewed and approved the American Centrifuge Lead Cascade Facility RP program, which complies with 10 CFR Part 20, "Standards for Protection against Radiation."

4.3.1 RP Program Implementation

The organizational responsibilities and personal qualifications for the implementation of the RP program are addressed in Chapter 2 of this SER. The RPM (also called the site's radiation safety officer) will be responsible for annually reviewing the content and implementation of the ACP RP program. The RPM will be responsible for providing guidance and direction for establishment and implementation of the RP Program. The RPM has direct access to the Director, American Centrifuge Plant, and the Vice President, American Centrifuge. The Health Physics (HP) Group reports to the RPM and provides radiological protection support to the facility.

The HP Group will be staffed with trained individuals that will provide oversight and control of the technical aspects of the program elements that affect RP. The applicant used provisions contained in 4.3.3, 4.4.5, and 4.5.3.2 of American National Standard Institute/American Nuclear Society (ANSI/ANS) ANSI/ANS N3.1-1993 (ANSI/ANS, 1993) to develop the qualifications of RP personnel. The HP Group will be independent of operations because the RPM has direct access to the Director, American Centrifuge Plant, and the Vice President, American Centrifuge, and the Health Physics (HP) staff reports to the RPM.

The applicant will review the RP program content and implementation annually, in accordance with 10 CFR 20.1101(c). The RPM will be responsible for writing an annual report based on the review, which the ALARA Committee will review.

The staff reviewed the applicant's RP program implementation against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 4.4.1.3 and, based on its review, the staff finds that:

1. The design and implementation of the applicant's RP program meets the regulatory requirements of 10 CFR Part 20, Subpart B because the RP program is independent of operations, the RPM will review the RP program at least once per year, and the applicant has committed to ANSI/ANS N3.1-1993 (ANSI/ANS, 1993) qualified staff;

2. The applicant has outlined the RP program structure and has defined the responsibilities of key program personnel;

3. The applicant will staff the RP program with suitably trained people in accordance with ANSI/ANS N3.1-1993 (ANSI/ANS, 1993) and has committed to providing sufficient resources to implement the program;

4. The RP program will be independent of operations, because the RPM has direct access to the Director, American Centrifuge Plant, and the Vice President, American Centrifuge, and the HP staff reports to the RPM; and

5. The applicant will perform an annual review of the RP program as required by 10 CFR 20.1101(c).

4.3.2 ALARA Program

In Section 4.2 of the LA (USEC, 2006b), the applicant describes the program that will maintain radiation doses ALARA. The program will be implemented in accordance with 10 CFR 20.1101 through written procedures.

The applicant states that the principles of the program will be to maintain personnel radiation exposures and the release of radioactive effluents in accordance with the ALARA principle, and to ensure that no individual receives a radiation dose in excess of any regulatory limit. The responsibility for establishing these principles rests with the Vice President, American Centrifuge. The Director, American Centrifuge Plant, has the overall responsibility and authority for the ALARA program and the RPM will be responsible for establishing and implementing the ALARA program.

The applicant will establish an ALARA Committee, which will be an independent advisory group to the Director, American Centrifuge Plant, and the Plant Safety Review Committee. The ALARA Committee will monitor selected operational RP issues, advise ACP management on RP concerns, and review proposed designs, work practices, selected suggestions, and selected projects regarding to contamination control and/or ALARA. The ALARA Committee will also establishs annual contamination control and exposure goals. Matters that have or may have an impact on contamination control and/or ALARA will be reviewed by the ALARA Committee, including evaluation of HP audit results, reports of radiation levels, contamination levels, employee exposures, and effluent releases. Technologies for selected job tasks, work practices and completed tasks related to contamination control or ALARA, RP violations, lessons learned, trends, and environmental monitoring reports will also be reviewed. ALARA Committee recommendations will be documented and tracked for completion. The ALARA Committee's duties and responsibilities are sufficient to facilitate interaction between RP and operations personnel because both operations and radiation safety management will be participating on the ALARA Committee and will make joint recommendations for improvements to Director, American Centrifuge Plant.

The ALARA Committee will consist of persons from various functional disciplines of the ACP. The committee will meet at least annually and as directed by the chairperson (RPM). A quorum consists of five standing committee members or their alternates.

The staff reviewed the applicant's ALARA program against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 4.4.2.3 and, based on its review, the staff finds that:

1. The applicant has committed to a comprehensive, effective, and written ALARA program;

2. The applicant will prepare policies and procedures to ensure that occupational exposures will be maintained ALARA and that such exposures are consistent with the requirements of 10 CFR 20.1101;

3. The applicant has outlined specific ALARA program goals, established an ALARA program organization and structure, and has committed to establish written procedures for implementation in the facility design and operations;

4. The applicant has committed to establishing an ALARA Committee with sufficient staff, resources, and clear responsibilities to ensure that the occupational radiation exposure dose limits specified in 10 CFR Part 20 are not exceeded under normal operations;

5. The applicant will use the ALARA program as a mechanism to facilitate interaction between RP and operations because both operations and radiation safety management

will be participating on the ALARA Committee and making joint recommendations for improvements to the Director, American Centrifuge Plant; and

6. The applicant will regularly review and revise the ALARA program goals and objectives based on new technologies and operating experience.

4.3.3 Organization and Personnel Qualifications

In Section 4.3 of the LA (USEC, 2006b), the applicant states that the organizational responsibilities and personal qualifications for implementation of the RP program are addressed in Section 2.1.3.3.1.1 of the LA (USEC, 2006b). The RPM directs the RP program and will be responsible for the implementation of the RP program. The RPM has direct access to the Director, American Centrifuge Plant, and Vice President, American Centrifuge, for radiological control matters, and reports to the Production Support Manager, which provides independence from operations.

HP Technicians and their managers will assist and guide workers in the radiological aspects of the job and will have the responsibility and authority to stop radiological work or mitigate the effect of an activity if they suspect the initiation or continued performance of a job, evolution, or test will result in a violation of approved RP requirements.

In Section 2.1.3.3.1.1 of the LA (USEC, 2006b), the applicant states that the RPM will have, as a minimum, a bachelor's degree in engineering, HP, RP, or the physical sciences or equivalent technical experience, and four years experience in RP including 6 months at a uranium processing plant. The staff finds the applicant's proposed minimum qualification for the RPM is sufficient for the type of radiation risks for this facility and therefore is acceptable.

HP Technicians will be trained in accordance with an approved qualification standard that will meet ANSI/ANS N3.1-1993 (ANSI/ANS, 1993) and will be re-qualified every 2 years. HP Technicians will be trained in accordance with an approved qualification standard developed by the applicant, which used provisions contained in 4.3.3, 4.4.5, and 4.5.3.2 of ANSI/ANS N3.1-1993 (ANSI/ANS, 1993) and will be re-qualified every two years.

The staff reviewed the applicant's organization and personnel qualifications against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 4.4.3.3 and, based on its review, the staff finds that:

1. The applicant has identified appropriate qualifications for the RPM and RP staff and has identified their authority and responsibilities;

2. The applicant has established clear organizational relationships for the RP program and other line managers;

3. The applicant will appoint a suitably trained RPM, who has direct access to the Director, American Centrifuge Plant, and Vice President, American Centrifuge, for radiological control matters. The RPM will be skilled in the interpretation of data and regulations pertinent to RP, will be familiar with the operation of the facility and RP concerns of the site, will be used as a resource in radiation safety management decisions, and will be responsible for establishing and implementing the RP program;

4. The applicant will assign responsibility to the RP program staff for implementation of the RP program functions; and

5. The applicant has described the minimum training requirements and qualifications for the RP staff.

4.3.4 Written Procedures

In Section 4.4 of the LA (USEC, 2006b), the applicant describes the implementation of the RP program using written procedures which the applicant states will be prepared in accordance with the requirements of 10 CFR Part 20. The applicant describes in Section 11.4 of the LA (USEC, 2006b) a management measures program in which the procedures will be prepared, maintained, and made available to appropriate personnel at the facility. See Chapter 11 of this SER for additional information on the management measures program.

The applicant states that RWPs will be used for activities in Contamination Areas, High Contamination Areas, Airborne Radioactivity Areas (ARAs), Radiation Areas (RAs), High Radiation Areas (HRAs), and other areas as required by the HP group. HP personnel will be trained in accordance with an approved qualification standard developed by the applicant, which used provisions contained in 4.3.3, 4.4.5, and 4.5.3.2 of ANSI/ANS N3.1-1993 (ANSI/ANS, 1993). These HP personnel will be authorized to approve, issue, update, revise, and close RWPs. RWPs may be issued for any period of time up to 1 year, may be closed at any time by HP, and will be normally closed upon job completion. For short duration, non-complex tasks, continuous HP coverage may be used in lieu of an RWP when approved by the RPM.

The staff reviewed the applicant's commitment to written procedures against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 4.4.4.3 and, based on its review, the staff finds that:

- Qualified HP personnel will prepare written RP procedures to carry out activities related to the RP program;

- The applicant specified how the RP procedures will be prepared, authorized, approved, and distributed; and

- The applicant will use approved RWPs for activities involving licensed material that are not covered by written RP procedures.

4.3.5 Training

In Section 4.5 of the LA (USEC, 2006b), the applicant describes the RP training program. The applicant will require both workers and visitors who enter controlled areas of the ACP where radioactive material may be encountered to be trained to the appropriate level for a visitor, or a general employee, or a radiological worker, or a HP technician, commensurate with the hazards as defined in 10 CFR Part 19 and Part 20.

In Section 11.3.1.3 of the LA (USEC, 2006b), the applicant describes a Radiological Worker Training program that outlines the requirements of 10 CFR 19.11 and 19.12 and the workers' responsibilities under the applicant's RP program. The applicant will use Regulatory Guide 8.13

(NRC, 1994) as part of the training program.

The applicant has committed to: (a) keep workers informed of the location or use of radioactive material; (b) instruct workers in the safety actions regarding exposure to radiation or radioactive material; (c) instruct workers as appropriate in the requirements of NRC regulations and its NRC license; (d) instruct the workers on their responsibility to report to the licensee any condition which may lead to or cause a violation of its license; (e) instruct workers on the appropriate response to warnings made in the event of any unusual occurrence or malfunctions of equipment; and (f) advise the workers that they may request exposure reports pursuant to 10 CFR 19.13. This meets the requirements of 10 CFR 19.12(a).

Radiological worker training will be required biannually for personnel having unescorted access to the Restricted Area. HP Technicians will be re-qualified every 2 years, which may include passing a written examination. The applicant will review and evaluate the RP program effectiveness and adequacy every 2 years.

Regulatory Guide 8.13 (NRC, 1994) is referenced in NUREG-1520 (NRC, 2002) Sections 4.4.2.2 and 4.4.5.2. This regulatory guide was used by the applicant and is acceptable because it pertains to female employees who could possibly be pregnant.

The staff reviewed the applicant's training program against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 4.4.5.3 and, based on its review, the staff finds that:

1. The applicant has designed and will implement an employee RP training program that complies with the requirements of 10 CFR Parts 19 and 20;

2. The applicant will provide appropriate training to all personnel and visitors entering restricted areas, or provide trained escorts;

3. The applicant will provide training based on the potential radiological health risks associated with that employee's work responsibilities;

4. The applicant will incorporate in the RP training program the provisions of 10 CFR 19.12 and will incorporate the appropriate topics;

5. The applicant will review the RP training program and conduct refresher training at least every 2 years; and

6. The applicant will evaluate the effectiveness and adequacy of the training program curriculum and instructors.

4.3.6 Ventilation and Respiratory Protection Programs

In Section 4.6 of the LA (USEC, 2006b), the applicant describes the ACP ventilation and respiratory protection programs. Design features of the ventilation systems are described in Chapter 1 of the LA (USEC, 2006b). The building ventilation systems will be primarily designed to maintain the building environment required for proper operation of process and associated equipment. Administrative guidance will require shutdown of the ventilation systems following detection of a uranium hexafluoride (UF_6) release.

The applicant may use fixed and portable ventilation units when the unprotected worker could potentially exceed 0.8 Derived Air Concentration (DAC)-hours of exposure. The local ventilation units will be equipped with high efficiency-particulate air (HEPA) filters.

The portable HEPA filter units' differential pressure will be checked per operating procedure, and the range will be based on the manufacturer's recommendations, or as specified in the technical design basis. As indicated in Section 4.6.1 and Section 1.4.4 of the LA (USEC, 2006b), all HEPA filter systems will be efficiency tested in accordance with American Society of Mechanical Engineers (ASME) N510-1989, "Testing of Nuclear Air-Treatment Systems" (ASME, 1989b). For those systems not designed in accordance with ASME N509-1989, "Nuclear Power Plant Air-Cleaning Units and Components" (ASME, 1989a), ASME N510-1989 will be met as testing guidance.

The applicant will maintain the average air velocity above 100 feet per minute through openings in uranium sampling and handling hoods containing readily dispersible uranium, and this velocity will be checked at least annually. Radiological containments that could generate airborne radioactivity will be maintained at a negative differential pressure, if used.

The applicant has committed to having a Respiratory Protection program that will meet the requirements of 29 CFR 1910.134 and 10 CFR Part 20 for use, issuance, training, records maintenance, and qualifications for respiratory protection users. Procedures will be prepared to comply with 10 CFR 20.1703(c)(4). RWPs will be used to specify respiratory protection for radiological protection purposes and will be considered for activities where an individual may be exposed to soluble uranium that may exceed 0.8 DAC-hours or an intake of 1 mg of soluble uranium during a shift.

Respiratory protection is considered during: (a) entry into posted ARAs; (b) breach of contaminated systems or components; (c) work in areas or on equipment with removable contamination levels greater than 100 times the levels in Table 4.6-1 of the LA (USEC, 2006b); and (d) work on contaminated surfaces with the potential to generate airborne radioactivity. The RPM may authorize situations where respiratory protection use is not practical because of physical limitations, such as heat stress, or is not in accordance with ALARA. Stay time controls and continuous workplace airborne monitoring will be provided in these situations.

The applicant commits to ASME N509-1989 (ASME, 1989a). New and existing fixed HEPA filter systems needed to ensure compliance with release limits or to control worker radiation exposure satisfy the provisions of this standard with the following exceptions/clarifications:

- Section 5.2 - Do not satisfy; No credit is taken for absorbers;
- Section 5.5 - Do not satisfy requirements for air heaters;
- Section 8.0 - Quality assurance requirements for applicable systems are identified in the Quality Assurance Program Description (QAPD) (USEC, 2006c);
- Appendix A - Do not sample absorbents;
- Appendix B - Do not use allowable leakage guidance;

- Appendix C - This appendix is used as guidance only; and

- Appendix D - The manifold qualification program uses this appendix as guidance only.

For the reference to this standard, see Section 4.6.1 of the LA (USEC, 2006b).

The applicant commits to ASME N510-1989 (ASME, 1989b). New and existing fixed HEPA filter systems that satisfy the requirements of ASME N509-1989 (ASME, 1989a) and are needed to ensure compliance with release limits or to control worker radiation exposure satisfy the provisions of this standard with the following exceptions/clarifications:

1. Section 6.0 - Only satisfy this section for new seal-welded duct systems or for connections to a system where this section has been previously applied; and

2. Section 7.0 - Do not use guidance for monitoring frame pressure leak tests.

Existing fixed HEPA filter systems that do not meet the requirements of ASME N509-1989 (ASME, 1989a) are tested using the requirements of this standard or another industry accepted standard as guidance only. For the reference to this standard, see Section 4.6.1 of the LA (USEC, 2006b).

The staff reviewed the applicant's ventilation and respiratory protection programs against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 4.4.6.3 and, based on its review, the staff finds for radiation safety purposes that:

1. The applicant will install appropriately sized ventilation and containment systems;

2. The applicant has described management measures to ensure that the ventilation and containment systems designated as IROFS operate when required, and are within their design specifications in accordance with the commitment to use ASME N509-1989, (ASME, 1989a) and ASME N510-1989 (ASME, 1989b);

3. The applicant has described the design criteria for the ventilation and containment systems in sufficient detail to demonstrate compliance with 10 CFR Part 20, Subpart H. This is based on the applicant's commitment to ASME N509-1989, (ASME, 1989a) and ASME N510-1989 (ASME, 1989b);

4. The applicant has described the frequency and types of tests to measure ventilation and containment systems performance and actions to be taken when the acceptance criteria are not satisfied;

5. The applicant has established a respiratory protection program that meets the requirements of 10 CFR Part 20, Subpart H;

6. The applicant will prepare written procedures for the selection, use, issuance, maintenance, testing, record keeping and training of personnel for individual respiratory protection equipment;

7. The applicant will revise procedures of use of respiratory protection equipment as

appropriate, based on the applicant's commitment to meet the requirements of 10 CFR 20.1703; and

8. The applicant will maintain records of its respiratory protection program, based on the applicant's commitment to meet the requirements of 10 CFR 20.1703.

4.3.7 Radiation Survey and Monitoring Programs

In Section 4.7 of the LA (USEC, 2006b), the applicant describes its radiation survey and monitoring programs, which are based on the requirements of 10 CFR Part 20, Subpart F, and ALARA principles and implemented through written procedures.

Surveys performed will consist of routine, work support, and material release surveys. Qualified HP Technicians will perform surveys for the purposes of establishing personnel protection equipment or for posting requirements.

The routine survey program will determine workplace radiological conditions, effectiveness of contamination control measures, and ensure identification and posting of radiological hazards. Survey areas will be categorized and scheduled commensurate with their relative radiological hazard and contamination potential. Survey frequencies will be based on area occupancy, potential for spread of contamination, and process knowledge. Routine contamination survey frequencies are listed in Table 4.7-1 of the LA (USEC, 2006b).

The applicant restricts the release of materials, equipment, and other items for unrestricted use if removable surface contamination levels equal or exceed those specified in Table 4.6-1 (USEC, 2006b). These levels are consistent with the guidelines prescribed in NRC's Branch Technical Position entitled "Guidelines for Decontamination of Facilities and Equipment prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material" (NRC, 1993a).

Large areas with removable contamination levels on accessible surfaces exceeding the levels specified in Table 4.6-1 of the LA (USEC, 2006b) will be posted as a contamination area or high contamination area, and actions will be taken to determine the source.

The personnel monitoring program includes an administrative control level (ACL) of 500 mrem per year Total Effective Dose Equivalent per person, an intake limit for soluble uranium of 10 mg per week, personnel dosimeters to measure the external exposure of personnel, analysis of personnel exposure, maintenance of exposure records, and a network of fixed nuclear accident dosimeters situated in ACP areas requiring a criticality accident alarm system, which also serve as area monitors. Personnel dosimeters will be also evaluated for neutron dose.

The applicant will have personnel requiring radiation exposure monitoring per 10 CFR 20.1502(a) wear beta-gamma sensitive dosimeters. The applicant will have dosimeters processed and evaluated by a processor holding current National Voluntary Laboratory Accreditation Program (NVLAP) accreditation from the National Institute of Standards and Technology (NIST). The dosimeters will be exchanged at least quarterly (+/-2 weeks) unless authorized in writing by the RPM. Self-reading or alarming dosimeters will be used for entry into HRAs or very HRAs. Self-reading or alarming dosimeters used by the applicant will not be used to assign radiation dose to workers, and accordingly, do not require NVLAP accreditation.

The RPM will perform an evaluation if an individual exceeds 50 percent of the ACL during a calendar quarter or the ACL in the calendar year. The RPM's evaluation, specified by procedure, will determine the cause, assess the exposure, and document the results.

HP personnel will review external dosimetry results to determine unusual trends or exposures. If the exposure status of an individual is uncertain, the individual will be removed from further exposure until the cause is determined and management is advised of any special controls or restrictions as a result.

The RPM will refer to the Corrective Action program any incident that results in airborne occupational exposures to radiation exceeding its administrative dose limits in 10 CFR Part 20 Appendix B, or the performance requirements of 10 CFR 70.61.

The applicant will use Regulatory Guide 8.34 (NRC,1992b) to comply with the requirements of 10 CFR 20.1202 for summation of external and internal radiation exposures, and the site will submit personnel monitoring information to the Radiation Exposure Information Reporting System (REIRS) based on the personnel exposure database, in compliance with the requirements of 10 CFR 20.2206.

The applicant will have personnel who have the potential to receive intakes resulting in a committed effective dose equivalent (CEDE) greater than or equal to 0.1 rem CEDE in a year, or intakes of 1 mg of soluble uranium per week, participate in the routine bioassay program. A description of the program and analytical methods employed will be given in Table 4.7-2 of the LA (USEC, 2006b). Bioassay sample frequencies and administrative action levels are given in Table 4.7-3 of the LA (USEC, 2006b). The applicant used data contained in Tables 2-1 and 2-2 of Federal Guidance Report No. 11, (EPA, 1988) to calculate dose conversion factors for radionuclides of concern. This data is also used to calculate the DAC listed in Table 4.7-4 of the LA (USEC, 2006b).

HP personnel will review urinalysis results to determine unusual trends. If bioassay results exceed ACLs, or as determined by HP, personnel will participate in follow-up bioassay monitoring. When intakes are confirmed or suspected to exceed 1 mg of soluble uranium per week, special bioassay studies will be performed as necessary and the applicant will conduct an investigation.

The applicant will perform routine air sampling, using the guidance of Regulatory Guide 8.25 (NRC,1992a), in areas where airborne radioactivity concentrations may exceed 10 percent of the DAC listed in Table 4.7-4 of the LA (USEC, 2006b), averaged over 8 hours. Airborne radioactivity posting levels are also listed in Table 4.7-4 of the LA (USEC, 2006b). The applicant will perform investigations and special bioassay sampling when air samples exceed 0.8 DAC-hours.

The applicant will have low-volume, high-volume, and lapel air samplers for job coverage and general area sampling. Samples will be routinely allowed to decay for a minimum of 3 days because of radon and radon daughter products. Air sample flow measurement devices will be calibrated at least annually, and lapel samplers will be calibrated in accordance with a written procedure.

The applicant will disposition deficiencies and provide the appropriate reports to NRC, in

accordance with the requirements of 10 CFR Parts 19, 20, and 70 associated with surveys and the monitoring program or results that exceed the applicant's ACL in accordance with the QAPD (USEC, 2006c) and Corrective Action Program, which is outlined in Section 11.6 of the LA (USEC, 2006b).

RP instrumentation and sensitivities are listed in Table 4.8-2 of the LA (USEC, 2006b). The instruments will be selected to measure the types and energies of radiation encountered with gas centrifuge enrichment operations. The RPM will be responsible for maintaining adequate quantities of calibrated radiation detection and measurement instruments.

In accordance with all sections of ANSI N323-1978, "Radiation Protection Instrumentation Test and Calibration" (ANSI, 1978), except for Sections 4.6, and 5.1.3, the applicant will inspect, maintain, and calibrate portable radiation detection and measurement instruments at least annually or the instrument will be removed from service. Section 4.6, "Periodic Performance Test," and Section 5.1.3, "Calibration Standards Where No National or Derived Standard Exists," are not applicable for types of instruments used by the applicant. The applicant will calibrate the instrument after any maintenance, modification, or repair deemed likely to affect operation before being returned to service. Calibration sources and equipment used for dose rate instruments will be within 5 percent of the stated value and have documented traceability links to the NIST large area uranium slab sources will be certified to 10 percent by NIST. Calibration sources for contamination monitoring equipment will be within 20 percent for activity and 10 percent for surface emission rate. The applicant will source check portable HP instruments that will be in use but do not have a built in automatic functional test features daily, or before using the instruments if not used on a daily basis. The applicant will check instruments with the automatic functional test feature that will be used once a week.

The applicant will require personnel exiting CAs and contamination control zones to monitor themselves for contamination after removing their protective clothing and before leaving the step-off pad area. Equipment and materials will be monitored and decontaminated as necessary before removal, or will be controlled as radioactive material.

The applicant will test sealed sources containing more than 100 microcuries (µCi) of beta and/or gamma emitting material or more than 10 µCi of alpha emitting material, other than ^{3}H, with a half-life greater than 30 days and in any form other than gas, for leakage and/or contamination at intervals not to exceed 6 months. The applicant will test sealed plutonium alpha sources containing 0.1 µCi or more of plutonium when in use at least every 3 months. This is consistent with the guidance contained in "License Condition for Leak-Testing Sealed Byproduct Material Sources" (NRC, 1993b); "License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters" (NRC, 1993c); and "License Condition for Leak-Testing Sealed Uranium Sources" (NRC, 1993d).

Posting criteria for restricted areas of the ACP are listed in Table 4.8-1 of the LA (USEC, 2006b). The applicant will provide radiological control by controlling access to areas where radioactive material may be encountered, by requiring that each person who enters those areas or facilities receives the appropriate level of radiological worker training, and by requiring personnel monitoring before exiting established step-off pad areas.

Regulatory Guide 8.25, (NRC,1992a) is referenced in NUREG-1520 (NRC, 2002) Section 4.4.7.2. The applicant will use the guidance of Regulatory Guide 8.25 (NRC,1992a) in areas

where airborne radioactivity concentrations may exceed 10 percent of the DAC listed in Table 4.7-4 of the LA (USEC, 2006b), averaged over 8 hours. Airborne radioactivity posting levels are also listed in Table 4.7-4 of the LA (USEC, 2006b). The applicant will perform investigations and special bioassay sampling when air samples exceed 0.8 DAC-hours.

Regulatory Guide 8.34 (NRC,1992b) is referenced in NUREG-1520 (NRC, 2002) Section 4.4.7.2. The applicant will use Regulatory Guide 8.34 (NRC,1992b) to comply with the requirements of 10 CFR 20.1202 for summation of external and internal radiation exposures, and the applicant will submit personnel monitoring information to the REIRS based on the personnel exposure database, in compliance with the requirements of 10 CFR 20.2206.

The applicant will use ANSI N323-1978 (ANSI, 1978) for RP instrumentation test and calibration, except for Sections 4.6, and 5.1.3. Section 4.6 “Periodic Performance Test,” and Section 5.1.3, “Calibration Standards Where No National or Derived Standard Exists,” are not applicable for the types of instruments the applicant will use. For the reference to this standard, see Section 4.8.4 of the LA (USEC, 2006b).

The applicant used data contained in Tables 2-1 and 2-2 of the Federal Guidance Report No. 11 (FGR, 1988) to calculate dose conversion factors for radionuclide of concern. This data is also used to calculate the DACs listed in Table 4.7-4. For the reference to this guidance document, see Section 4.7.4 of the LA (USEC, 2006b).

The staff reviewed the applicant’s radiation surveys and monitoring program against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 4.4.7.3 and, based on its review, the staff finds that:

1. The applicant’s monitoring program is consistent with the requirements of 10 CFR Part 20, Subpart F;

2. The applicant will prepare written procedures for the radiation survey and monitoring program sufficient to meet the requirements of 10 CFR Part 20;

3. The applicant will design and implement a personnel monitoring program for external occupational radiation exposures;

4. The applicant will design and implement a personnel monitoring program for internal occupational radiation exposures, that will meet the requirements of 10 CFR 20.1201, 20.1204, and 20.1502(b);

5. The applicant will comply with the requirements of 10 CFR 20.1202 for summation of external and internal occupational radiation exposures, by using appropriate procedures, in accordance with the commitment to use Regulatory Guide 8.34 (NRC, 1992b);

6. The applicant will have an air sampling program, will conduct air surveys, and will have procedures to calibrate and maintain airborne sampling equipment in accordance with the manufactures’ recommendations in accordance with the commitment to use Regulatory Guide 8.25 (NRC, 1992a);
7. The applicant will implement additional procedures as may be required by 10 CFR Part

20 and the ISA Summary (USEC, 2006a) to control the concentration of airborne radioactive material;

8. The applicant will conduct a contamination survey program in areas of the facility identified in the ISA Summary (USEC, 2006a) most likely to be radiologically contaminated;

9. The applicant will implement its Corrective Action program when the results of personnel monitoring or contamination surveys exceed the applicant's administrative personnel contamination levels;

10. The applicant will implement its Corrective Action program when any incident resulting in airborne occupational exposures to radiation that exceeds its administrative limits, the dose limits in 10 CFR Part 20, Appendix B, or the performance requirements of 10 CFR 70.61;

11. The applicant will use appropriate survey equipment and instrumentation and will calibrate the equipment in accordance with the manufacturers' recommendations in accordance with the commitment to use ANSI N323-1978 (ANSI, 1978);

12. The applicant will use NRC Branch Technical Position "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material" (NRC, 1993a) to ensure that materials leaving restricted areas will have safe levels of contamination;

13. The applicant will perform leak-testing of all sealed sources in an acceptable manner, because it is consistent with the guidance contained in "License Condition for Leak-Testing Sealed Byproduct Material Sources" (NRC, 1993b); "License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters" (NRC, 1993c); and "License Condition for Leak-Testing Sealed Uranium Sources" (NRC, 1993d);

14. The applicant has described an acceptable Access Control program; and

15. The applicant will have a Radiation Exposure Reporting program that meets the requirements of 10 CFR Parts 19 and 20.

4.3.8 Additional Program Requirements

The applicant will maintain RP program records that follow the guidance of ANSI N13.6-1999 (ANSI,1999) and has committed to procedures to retain its records as required by 10 CFR 20.2101 through 20.2106.

Reports and notifications of RP issues will be made as required by 10 CFR Part 20, Subpart M, 10 CFR 30.50, 10 CFR 40.60, and 10 CFR 70.74.

The applicant will submit personnel monitoring information to NRC (REIRS) based on the personnel exposure database, in compliance with the requirements of 10 CFR 20.2206(b).

The staff reviewed the applicant's additional program commitments against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 4.4.8.3 and, based on its review, the staff finds that:

1. The applicant will maintain records of the RP program, radiation survey results, and results of its Corrective Action program referrals, RWPs' and planned special exposures;

2. The applicant will establish a program to report to NRC, within the time specified in 10 CFR 20.2202 and 10 CFR 70.74, any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR Part 20;

3. The applicant will submit to NRC an annual report required by 10 CFR 2206(b); and

4. The applicant will refer to its Corrective Action program occupational exposures that exceed the dose limits in 10 CFR Part 20, Appendix B, that are required to be reported per 10 CFR 70.74, and report the results to NRC.

The applicant has requested an exemption from 10 CFR Part 20 requirements related to posting and labeling each container of licensed material. 10 CFR 20.1904 requires that each container of licensed material bear a durable, clearly visible label such that the radionuclide(s) present, the quantity of radioactivity, radiation levels, kinds of materials, mass, and enrichment be identified. The applicant states that it will be impractical to label each and every container in restricted areas, and will have one sign posted stating that every container may contain radioactive material in these areas. The applicant will perform a survey when containers will be removed from contaminated or potentially contaminated areas to prevent spread of contamination. The applicant also requested that the UF_6 feed, product, and depleted uranium cylinders not be labeled because they will be readily identifiable because of their size and unique construction. The applicant also states that UF_6 cylinders will be constantly attended by qualified radiological workers during movement.

Under 10 CFR 20.2301, the Commission may grant exemptions from the requirements of the regulations, if it determines that the request will be authorized by law and will not result in undue hazard to life or property. Also, 10 CFR 20.1905(c) already exempts containers from 10 CFR 20.1904, if the containers are attended by an individual who takes the precautions necessary to prevent the exposure of individuals in excess of the limits established. The staff agrees that it would be impractical to label each and every container in restricted areas at this facility because of the large number of potential containers. Labeling each container may also reduce radiation safety by desensitizing the worker to radiation warning signs. Since there is no statutory provision prohibiting the granting of this exemption, the staff concludes that the request is authorized by law. Also, the exemption request is consistent with those approved previously at the gaseous diffusion plants and other fuel cycle facilities. Experience at facilities that have received the exemption from the labeling requirement demonstrates that the applicant's request will provide an equivalent amount of safety, and will not result in an undue hazard to individuals. Accordingly, the staff finds that the request will not be an undue hazard to life or property. Therefore, exemption to the requirements of 10 CFR 20.1904 is recommended.

The applicant has also requested that in lieu of the requirements of 10 CFR 20.1601(a), each HRA will be conspicuously posted "Caution, High Radiation Area," and entrance into the area will be controlled by an RWP. The applicant will also implement physical and administrative

controls to prevent inadvertent or unauthorized access to HRAs and very HRAs.

An applicant may apply to the Commission under 10 CFR 20.1601(c) for approval of alternative methods for controlling access to HRAs. The staff finds the applicant's use of conspicuously posted signs with the applicant's RWP program an acceptable alternative, pursuant to 10 CFR 20.1601(c).

4.4 EVALUATION FINDINGS

The applicant has established and will maintain an acceptable RP program that includes:

1. An effective documented program to ensure that occupational radiological exposures are ALARA;

2. An organization with adequate qualification requirements for the RP personnel;

3. Approved, written RP procedures and RWPs for RP activities;

4. RP training for all personnel who have access to restricted areas;

5. A program that will control airborne concentrations of radioactive materials with engineering controls and respiratory protection;

6. A radiation survey and monitoring program that will include requirements for controlling radiological contamination within the facility and monitoring of external and internal radiation exposures; and

7. Other programs to maintain records, report to NRC in accordance with 10 CFR Parts 20 and 70, and correct for upsets at the facility.

NRC staff concludes that the applicant's RP program is adequate and meets the requirements of 10 CFR Parts 19, 20 and 70. Conformance to the LA (USEC, 2006b) will ensure safe operation.

4.5 REFERENCES

(ANSI, 1978) American National Standard Institute (ANSI). N323-1978, "Radiation Protection Instrumentation Test and Calibration," 1978.

(ANSI,1999) American National Standard Institute (ANSI). N13.6-1999, "Practice for Occupational Radiation Exposure Records Systems," 1999.

(ANSI/ANS, 1993) American National Standard Institute/American Nuclear Society (ANSI/ANS). N3.1-1993, "Selection, Qualification and Training of Personnel for Nuclear Power Plants," 1993.

(ASME, 1989a) American Society of Mechanical Engineers (ASME). N509-1989, "Nuclear Power Plant Air-Cleaning Units and Components," 1989.

(ASME, 1989b) American Society of Mechanical Engineers (ASME). N510-1989, “Testing of Nuclear Air-Treatment Systems,” 1989.

(EPA, 1988) U.S. Environmental Protection Agency (EPA). Federal Guidance Report No. 11. EPA 520/1-88-020, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September, 1988.

(NRC,1992a) U.S. Nuclear Regulatory Commission (NRC). Regulatory Guide 8.25, “Air Sampling In The Workplace,” June, 1992.

(NRC,1992b) U.S. Nuclear Regulatory Commission (NRC). Regulatory Guide 8.34, ”Monitoring Criteria and Methods to Calculate Occupational Radiation Dose,” July, 1992.

(NRC, 1993a) U.S. Nuclear Regulatory Commission (NRC). Branch Technical Position, “Guidelines for Decontamination of Facilities and Equipment prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material,” April, 1993.

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

(NRC, 1993c) U.S. Nuclear Regulatory Commission (NRC). Branch Technical Position, “License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters,” 1993.

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

(NRC, 1994) U.S. Nuclear Regulatory Commission (NRC). Regulatory Guide 8.13, “Instructions Concerning Prenatal Radiation Exposure,” 1994.

(NRC, 1999) U.S. Nuclear Regulatory Commission (NRC). “Compliance Evaluation Report for USEC Portsmouth Gaseous Diffusion Plant,” January, 1999.

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

(USEC, 2006a) USEC Inc. (USEC). “Integrated Safety Analysis Summary for the American Centrifuge Plant in Piketon, Ohio,” Revision 14, August 2006.

(USEC, 2006b) USEC Inc. (USEC). “License Application for the American Centrifuge Plant in Piketon, Ohio,” Revision 18, September 2006.

(USEC, 2006c) USEC Inc. (USEC). “Quality Assurance Program Description for the American Centrifuge Plant in Piketon, Ohio,” Revision 3, August 2006.

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.

The NCS programmatic review will determine whether: (1) the applicant has provided for the appropriate management of the NCS program; (2) the applicant has identified and committed to the responsibilities and authorities of individuals for developing and implementing the NCS program; (3) the facility management measures described in 10 CFR 70.62 have been committed to and will support implementing and maintaining the NCS program; and (4) an adequate NCS program is described, which includes identifying and committing to the NCS methods and NCS technical practices used to ensure the safe operation of the facility, as required by Part 70. This included review of the applicant's criticality code validation report to determine whether its use of calculational methods provides assurance that processes will be subcritical under normal and credible abnormal conditions, as specified in 10 CFR 70.61(d).

The NCS ISA review was performed to determine whether: (1) the ISA program is acceptable for NCS; (2) the ISA has been acceptably performed and will be maintained for NCS; and (3) the ISA Summary contains necessary information, such that the NCS accident sequences are "highly unlikely."

5.1 REGULATORY REQUIREMENTS

The NCS review of the applicant's NCS program should verify if the information the applicant provided meets the requirements of 10 CFR 70.22 and 70.65, which, respectively, specify the general and additional content of a license application (LA). In addition, the NCS review should verify compliance with the regulatory requirements in 10 CFR 70.24, 70.52, 70.61, 70.62, 70.64, 70.65, 70.72, and 10 CFR Part 70, Appendix A.

The NCS review of the applicant's ISA program and ISA Summary should verify if the information the applicant provided meets the requirements of 10 CFR 70.62 and 70.65, which, respectively, specify: (1) the requirements for establishing and maintaining a safety program (10 CFR 70.62), including an ISA program that addresses NCS; (2) requirements for conducting and maintaining an ISA (10 CFR 70.62(c)) for NCS; and (3) requirements for the contents of an ISA Summary (10 CFR 70.65(b)) for NCS.

5.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The regulatory guidance applicable to the NCS review of the applicant's NCS program is contained in Chapter 5 of NUREG-1520 (NRC, 2002). The acceptance criteria are outlined in Sections 5.4.3.1, 5.4.3.2, 5.4.3.3, and 5.4.3.4 of NUREG-1520 (NRC, 2002). This includes the use of NRC NCS Regulatory Guide 3.71 (NRC, 2004b), which modified the use of the American National Standards Institute/American Nuclear Society (ANSI/ANS) Series-8 NCS standards

(ANSI/ANS, 1975, 1981, 1983a, 1983b, 1984, 1987a, 1987b, 1991, 1995, 1996a, 1996b, 1997a, 1997b, 1997c, and 1998) .

The acceptance criteria used for the NCS review of the applicant's ISA program and ISA Summary (USEC, 2006a) are outlined in Sections 3.4.3.1 and 3.4.3.2 of NUREG-1520 (NRC, 2002).

5.3 STAFF REVIEW AND ANALYSIS

This chapter addresses the staff's review of NCS, including management of the NCS program, organization and administration, management measures, methodologies and technical practices, use of national consensus standards, and criticality accident alarm system (CAAS) exemption as presented in the LA (USEC, 2006b).

The staff review of criticality code validation report and margin of subcriticality, NCS in the ISA, NCS in the Emergency Plan (EP), criticality related accident scenarios, and criticality related items relied on for safety (IROFS) is contained in Appendix C, "Nuclear Criticality Safety," of this Safety Evaluation Report (SER). The appendix evaluates information contained in the ISA Summary (USEC, 2006a) which is marked as "Export Controlled Information" by the applicant, and is designated by the U.S. Nuclear Regulatory Commission (NRC) as "Official Use Only-DOE/NOFORN."

NCS Risk Perspective for the American Centrifuge Plant (ACP)

Throughout many of the following sections, the staff's determination of the acceptability of the applicant's statements was informed by a consideration of the low risk of the majority of ACP operations. This is based on: (1) heavy reliance on passive geometry and limited mass in the majority of processes (except product withdrawal and uranium hexafluoride (UF_6) cylinder handling and storage); (2) heavy reliance on the integrity of passive equipment to ensure moderator control in most of the remaining, support processes; (3) limited handling of liquid UF_6 (i.e., limited to autoclaves and associated piping in the sampling, transfer, and blending facility), and no solution processing (responsible for the majority of historical criticality accidents) at the facility; and (4) processes being limited to less than 10 wt% ^{235}U (and product initially limited to less than 5wt% ^{235}U). Although there are parts of the facility that involve large quantities of enriched uranium in unfavorable geometry (e.g., product withdrawal and cylinder storage), the above factors permit a graded approach in NRC's review of safety in the facility overall.

5.3.1 Management of the NCS Program

5.3.1.1 Program Elements

Section 5.1.1 of the LA (USEC, 2006b) describes the major regulatory requirements to be met by the applicant's NCS Program. These requirements address the acceptance criterion to develop, implement, and maintain an NCS program listed in Section 5.4.3.1(1) of NUREG-1520 (NRC, 2002). These requirements include codification of essential program elements into procedures, identification of controls and barriers relied on to prevent criticality as IROFS, and meeting the baseline design criteria (BDC) of 10 CFR 70.64(a). Regarding the identification of criticality controls as IROFS, staff questioned whether this meant that all controls relied on to meet the double contingency principle (DCP) would be IROFS. The applicant clarified in its

response (USEC, 2005a) that each control relied on to meet the performance requirements of 10 CFR 70.61 would be designated an IROFS, but this did not mean that all controls relied on to meet the DCP would be IROFS. This is consistent with the requirements of 10 CFR Part 70, Subpart H, as elaborated in Interim Staff Guide ISG-03 (NRC, 2004a), which indicates that double contingency controls are only required to be IROFS if compliance with the DCP is used to demonstrate compliance with the performance requirements. The applicant's ISA approach uses a likelihood determination method based on failure frequency rather than compliance with the DCP, which is deterministic. The applicant's description of which controls are IROFS is therefore acceptable to the staff.

The program elements described here are described in greater detail in subsequent sections of the LA (USEC, 2006b) and will be discussed and evaluated further below.

5.3.1.2 Program Objectives

Section 5.1.2 of the LA (USEC, 2006b) enumerates the main duties to be performed by the NCS program. The list of duties is identical to that from Section 5.3.1 of NUREG-1520 (NRC, 2002), and implements the criterion to describe program objectives in Section 5.4.3.1(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable to the staff. These duties specifically include establishment of safe parameters and program procedures, and thereby meet the acceptance criterion in Section 5.4.3.1(3) of NUREG-1520 (NRC, 2002).

5.3.2 Organization and Administration

5.3.2.1 NCS Responsibilities

Section 5.2.1 of the LA (USEC, 2006b) describes the responsibilities of key NCS personnel. Those personnel having direct responsibility for NCS consist of the NCS Manager, Qualified NCS Engineers, and Senior NCS Engineers. The staff finds that the list of duties is consistent with the acceptance criteria in Section 5.4.3.1(4) of NUREG-1520 (NRC, 2002), that the applicant outlines the program structure and defines the responsibilities and authorities of key program personnel, and this is therefore acceptable to the staff. This also satisfies the acceptance criterion with regard to organizational positions and functional responsibilities in Section 5.4.3.2(1) of NUREG-1520 (NRC, 2002). The discussion of duties of the NCS Manager satisfies the acceptance criterion in Section 5.4.3.2(7) of NUREG-1520 (NRC, 2002).

5.3.2.2 NCS Staff Qualifications

Section 5.2.2 of the LA (USEC, 2006b) lists minimum educational and experience requirements for Qualified and Senior NCS Engineers (requirements for the NCS Manager are described in Section 5.2.1 of the LA (USEC, 2006b)). This satisfies the acceptance criterion with regard to experience and education in Section 5.4.3.2(1) of NUREG-1520 (NRC, 2002). The applicant provided further clarification in its response to Request for Information (RAI) NC-3 (USEC, 2005a) of what "equivalent technical experience" and "nuclear experience" meant in the qualifications for the NCS Manager. Although the NCS Manager is not required to have prior experience in NCS, this response states that the position is primarily administrative and the NCS Manager will not have the ability to make technical decisions unless this individual also meets the requirements for a Qualified or Senior NCS Engineer. Because Qualified and Senior NCS Engineers alone have the authority to perform the technical aspects of ensuring NCS, this

is acceptable to the staff.

The staff considered the educational requirements for Qualified and Senior NCS Engineers to be appropriate for the duties of these positions. With regard to experience requirements for Senior NCS Engineers, the applicant stated in its response to RAI NC-4 (USEC, 2005a) that the required 1-year term as a Qualified NCS Engineer is sufficient. Section 11.4.3.3 (9)(c) of NUREG-1520 (NRC, 2002) states that technical staff positions whose duties are critical to satisfy the performance requirements should have 3 years of experience. The applicant stated that the required 1-year term is only part of the qualification process, and the remaining components ensure the candidate has the requisite knowledge and skills to function as a Senior NCS Engineer. The applicant pointed out in a telephone call that it would typically take more than 1 year to complete the requirements for a Qualified NCS Engineer, such as performing at least four evaluations and completion of required coursework, as well as having familiarization with NCS by having at least 1 year of experience at an enriched uranium processing facility. Thus, most candidates for the position of Senior NCS Engineer will have substantially more than 1 year of experience (or at least familiarity) with NCS. Given the limited complexity of operations and criticality controls at the ACP, this is acceptable to the staff.

A commitment to describe organizational positions, experience, qualifications, and functional responsibilities as discussed in the acceptance criterion in Section 5.4.3.2(6) of NUREG-1520 (NRC, 2002), is not necessary, because the actual organizational and staff qualification requirements are listed in the LA (USEC, 2006b). For the same reason, a commitment to staff the NCS program with suitably trained personnel and provide sufficient resources, as described in the acceptance criteria in Section 5.4.3.2(8) of NUREG-1520 (NRC, 2002) is not needed.

5.3.3 Management Measures

5.3.3.1 Procedure Requirements

Section 5.3.1 of the LA (USEC, 2006b) states that operations to which NCS pertains will be governed by written procedures or work packages, and that NCS Evaluation (NCSE) controls that specify operator actions are incorporated into the procedures. Section 11.2.3 of the LA (USEC, 2006b) describes the Procedure Control program. As stated in Section 1.4.1 of the LA (USEC, 2006b), the applicant stated that it will follow ANSI/ANS-8.19-1996 (ANSI/ANS, 1996b) and ANSI/ANS-8.20-1991 (ANSI/ANS, 1991). Section 5.4.3.3(2) of NUREG-1520 (NRC, 2002) contains the acceptance criterion that the applicant will follow ANSI/ANS-8.19-1996 (ANSI/ANS, 1996b) as it pertains to procedures. In stating that it will meet ANSI/ANS-8.20-1991 (ANSI/ANS, 1991), the applicant has met the acceptance criteria regarding NCS training in Section 5.4.3.1(7) of NUREG-1520 (NRC, 2002). The statement that it will have procedures and will follow the standard is consistent with this guidance and is, therefore, acceptable to the staff.

5.3.3.2 Posting and Labeling Requirements

Section 5.3.2 of the LA (USEC, 2006b) contains requirements for posting and labeling of NCS administrative controls; these are intended as operator aids and therefore not applicable to engineered controls. This satisfies the acceptance criterion in Section 5.4.3.1(7) and Section 5.4.3.2(4) of NUREG-1520 (NRC, 2002) with regard to NCS postings. Postings are not required when in-hand procedures are used, as these contain the administrative requirements, and there

would be no value to posting them. There are other situations in which postings may not be required, and these are evaluated on a case-by-case basis. A list of these are provided in the applicant's response to RAI NC-7 (USEC, 2005c), which includes situations that are not limited to a well-defined area (making it difficult for postings to be placed in close proximity to workers) or controls that are incorporated into formalized tracking systems (e.g., maintenance) or are purely administrative (e.g., filling out logbooks, tagging out equipment). As postings are primarily used to remind operators of important NCS limits in areas where hands-on operations are performed, the staff finds these exceptions to be reasonable. In any event, the fissile material operations (FMO) manager and cognizant NCS engineer determine the need for postings on a case-by-case basis and document the reason when postings are not used. When postings are used, they will be printed in an appropriate size and placed in conspicuous locations. In its response to RAI NC-8 (USEC, 2005a), the applicant clarified that this meant the posting would be placed within conspicuous view of operators (either when performing operations or on entry to rooms or areas where operations are performed) and would be of sufficient size and simplicity to be readily legible to operators. The staff finds that this should ensure that postings of NCS limits will be effectively employed, and therefore this is acceptable to the staff.

5.3.3.3 Change Control

Section 5.3.3 of the LA (USEC, 2006b) addresses the configuration management program as applied to NCS, including defining the boundary of components necessary to ensure reliability and availability of NCS controls, and describes the change control process for changes that may affect NCS. This addresses the acceptance criteria on configuration management in Section 5.4.3.1(5) and (6) of NUREG-1520 (NRC, 2002). This also satisfies the acceptance criterion with regard to evaluating modifications to operations and control selection in Section 5.4.3.1(9) of NUREG-1520 (NRC, 2002). Changes that could potentially affect NCS will be reviewed by the NCS organization to determine whether the change affects the analysis and conclusions made in the applicable NCSE. Section 5.3.3 of the LA (USEC, 2006b) and the change pages (USEC, 2006e) states that "Changes that could establish new fissile material operations or affect established fissile material operations are reviewed by NCS." The staff finds that this practice should ensure that no changes that could impact the criticality safety basis of the facility are made without proper NCS review, and this is therefore acceptable to the staff. These change control requirements represents an acceptable alternative approach to the acceptance criterion in Section 5.4.3.4.7(2) of NUREG-1520 (NRC, 2002).

The applicant's measures to implement the facility change process satisfy the acceptance criteria in Section 5.4.3.4.7(6) of NUREG-1520 (NRC, 2002). Specifically, Section 5.3.3 of the LA (USEC, 2006b) implements the facility change process discussed in Section 5.4.3.4.7(6)(a) of NUREG-1520 (NRC, 2002), including the connection to the facility configuration management process as discussed in Section 5.4.3.4.7(6)(b) of NUREG-1520 (NRC, 2002). This includes determining whether prior NRC approval is required per 10 CFR 70.72, as discussed in Section 5.4.3.4.7(6)(c) of NUREG-1520 (NRC, 2002).

5.3.3.4 Operation Surveillance and Assessment

Section 5.3.4 of the LA (USEC, 2006b) contains a description of the various walk-throughs, self-assessments, and internal audits of operations and the NCS organization. The purpose of these walk-throughs, self-assessments, and audits is to ensure continued compliance both with NCS controls and NCS program requirements. The applicant stated that it would conduct at least annual walk-throughs of facility processes (although this may be increased due to performance trends or other concerns). Section 5.4.3.3(3)(b) of NUREG-1520 (NRC, 2002) contains the acceptance criterion that walk-throughs be done weekly such that all operating areas are reviewed at least every 2 weeks.

Section 5.4.3.3(3)(b) of NUREG-1520 (NRC, 2002) also states, however, that a graded approach may be used to justify an alternate walk-through schedule. The justification for the reduced schedule is the applicant's statements in Section 5.3.4 of the LA (USEC, 2006b) that front line managers (FLMs) are "provided additional training on NCS and response to NCS deficiencies" to enable them to provide real-time assessment of compliance with NCS requirements. This satisfies the policy discussion acceptance criterion in Section 5.4.3.2(5) of NUREG-1520 (NRC, 2002). This FLM training also satisfies the acceptance criterion in Section 5.4.3.3(1)(c) of NUREG-1520 (NRC, 2002). In addition, the applicant amended the LA (USEC, 2006b) to specify that although each area in the facility would be subject to a walk-through at least annually, the walk-throughs would be distributed throughout the year such that they would be conducted in some portion of the facility on an approximately monthly basis. Because of the limited reliance on administrative controls and simplicity of ACP operations, this lower frequency (than what is in NUREG-1520 (NRC, 2002)) is acceptable to the staff. In addition, during the on-site vertical slice reviews, the staff toured the unfinished facility and recognized that the facility in general (and the portion auxiliary to the cascade in particular) covered a relatively small area, such that NCS personnel would have ample opportunities to notice deficiencies more regularly than the annual frequency would imply.

Section 11.5 of the LA (USEC, 2006b) states that audits will be conducted of the various safety programs (including NCS) to ensure "comprehensive program oversight at least once every three years." Section 5.4.3.3(3)(c) of NUREG-1520 (NRC, 2002) contains the acceptance criterion that audits be done quarterly, such that all NCS aspects of management measures are reviewed every 2 years. However, in addition to the triennial audits, Section 5.3.4 of the LA (USEC, 2006b) states that FMO management will conduct annual self-assessments to ensure that NCS requirements are met in the field. In addition, internal audits of the NCS program will be conducted by the Quality Assurance Manager in accordance with Section 11.5 of the LA (USEC, 2006b). Given the limited complexity of operations and criticality controls, the staff considers the combination of annual self-assessments of field compliance of NCS and comprehensive program audits every 3 years appropriate. The aforementioned requirements regarding audits and assessments represents an acceptable alternative approach to the acceptance criteria in Section 5.4.3.4.7(1) of NUREG-1520 (NRC, 2002).

The acceptance criteria in Section 5.4.3.4.7(3) of NUREG-1520 (NRC, 2002) states that the applicant should commit to upgrade the NCS program to reflect changes in the ISA or new NCS methodologies and should modify the operating and maintenance procedures to reduce the likelihood of an inadvertent criticality. The staff considers this unnecessary, because: (1) the NRC Program is separate from and is not affected by changes to the ISA, and (2) 10 CFR 70.61 establishes the level of acceptable risk in performing the ISA, and there is no regulatory requirement to reduce likelihoods below "highly unlikely" for high-consequence events and "unlikely" for intermediate-consequence events. The acceptance criteria in Section 5.4.3.4.7(4)

of NUREG-1520 (NRC, 2002) states that the applicant should retain NCS records and document any corrective actions taken. The management measures discussed in Chapter 11 of this SER include configuration control (of which document control is a subset), and retention of such records is a prerequisite to being able to perform the audits and inspections discussed above. Therefore, a specific requirement to maintain NCS records is subsumed by a general requirement to retain safety-related records, and is not necessary.

5.3.4 Methodologies and Technical Practices

5.3.4.1 Adherence to ANSI/ANS Standards

Section 5.4.1 of the LA (USEC, 2006b) states that it will follow the national consensus standards ANSI/ANS-8.1-1998 (ANSI/ANS, 1998), ANSI/ANS-8.19-1996 (ANSI/ANS, 1996b), and ANSI/ANS-8.21-1995 (ANSI/ANS, 1995) subject to clarification below. These standards have been endorsed in Regulatory Guide 3.71, Rev. 1, "Nuclear Criticality Safety Standards for Fuel and Material Facilities" (NRC, 2004b), and this is therefore acceptable to the staff. Discussion of the other applicable ANSI/ANS-8 Series standards is also discussed below in Section 5.3.5 of this SER.

5.3.4.2 Process Evaluation and Approval

Section 5.4.2 of the LA (USEC, 2006b) discusses the performance of NCSEs. Processes involving uranium enriched to at least 1 wt% ^{235}U and at least 100 g ^{235}U will be evaluated for NCS before operations. These evaluations will include a hazard evaluation as well as a contingency analysis to demonstrate compliance with the DCP. This section also provides a list of cases in which an NCSE may not need to be performed in response to a facility operations' request for an NCS evaluation. In its response to RAI NC-12 (USEC, 2005a), the applicant provided a more detailed list of reasons that an NCSE may not be necessary. These reasons include: (1) the operation is obviously non-fissile; (2) the operation has not been funded; (3) the operation is within the scope of an existing NCSE; (4) the operation is non-fissile because of limited inventory (i.e., < 100 g ^{235}U) or enrichment (i.e., < 1 wt% ^{235}U); and (5) the operation cannot be shown to be doubly contingent or criticality shown to be "highly unlikely." The applicant stated (USEC, 2005c) that such a list was needed because, in its historical experience with the Gaseous Diffusion Plants (GDPs), operations had tended to be overly conservative in requesting NCSEs. This did not negate the fact, however, that all operations exceeding 1 wt% ^{235}U and 100 g ^{235}U would require NCSEs. The staff concluded that this approach is acceptable because the list of cases not needing NCSEs are reasonable and all operations exceeding 1wt% ^{235}U and 100 g ^{235}U will still be required to have an NCSE.

Section 5.4.2 of the LA (USEC, 2006b) also provides additional detail on how the applicant will meet the DCP. DCP will be ensured by establishing controls on one or more system parameters. Controls may be either passive or active engineered features, administrative controls, reliance on the natural and credible course of events, or other means that limit parameters within specified values. With regard to the use of the "natural and credible course of events," the applicant indicated (USEC, 2005a) that the natural and credible course of events would be inherent to facility operations without the need to credit additional passive, active, or administrative controls to maintain them. The applicant further stated in response to NRC's RAI NC-13 (USEC, 2005c) that the usage of the "natural and credible course of events" would be consistent with the intent ANSI/ANS-8.1-1998 (ANSI/ANS, 1998), and that it would just provide

more specificity as to how double contingency may be met. The use of such events is anticipated to be rare, because "in order the qualify as a natural and credible course of events, no controls will be necessary to maintain them" (USEC, 2005a), as confirmed during the on-site vertical slice review (which did not identify any instances of these events). The applicant stated that it will provide justification in NCSEs and the ISA Summary (USEC, 2006a) wherever the natural and credible course of events is credited for criticality safety (USEC, 2005a).

With respect to "other means," the language in the LA (USEC, 2006b) is also the same as that in ANSI/ANS-8.1-1998 (ANSI/ANS, 1998). The applicant also stated that it does not currently make use of "other means," and that if it does in the future, this will require pre-approval under 10 CFR 70.72, and Section 5.4.2 of the LA (USEC, 2006b) states that the use of the "natural and credible course of events" or "other means" would require prior NRC review and approval. The exceptions to this are: (1) reliance on other plant programs or management measures described in Chapter 11 of the LA (USEC, 2006b); (2) accident sequences that are not credible; and (3) sequences that do not result in a critical configuration even with loss of both double contingency controls. The first exception is acceptable to the staff because the use of such programs is discussed explicitly in Section 5.4.2 and Chapter 11 of the LA (USEC, 2006b). The second exception is acceptable to the staff because the determination of non-credibility must not be based on any facility features that are subject the licensee control (see SER Appendix C, Section C.2). The third exception is acceptable to the staff because there must be sufficient other controls in place in order to meet this condition. Therefore, the requirement to submit these cases (expected to be rare and subject to only these three exceptions), for prior NRC review and approval is acceptable to the staff.

With regard to RAI NC-14 regarding the process and/or criteria for determining when a change is sufficiently "unlikely" to meet the DCP, the applicant stated that the conclusion that a change in process conditions is unlikely would be made by the collective judgement of a number of individuals as part of the approval process (USEC, 2005c). This process is described in detail in Section 5.4.2 of the LA (USEC, 2006b). Most of the individuals involved in the initial design of the ACP had extensive experience with uranium enrichment, UF_6 handling, GDP operation, and GDP history at either the Portsmouth or Paducah GDP sites. Moreover, when available, the applicant stated that it would use actual operating history or failure frequencies to make the likelihood determination. Based on the involvement of multiple individuals and the statement that it would use available operating history or failure frequency data, this approach is acceptable to the staff.

With regard to RAI NC-15 regarding the definition of "items related to NCS" for criticality control purposes, the applicant stated that such items included factors that could be used to ensure criticality safety without reliance on any engineered or administrative controls (USEC, 2005a). These features could include the use of non-NCS programs (e.g., health physics, industrial safety) or inherent aspects of the process not tied to any specific controls. Two examples of inherent aspects of the process were provided: (1) a change in process conditions requiring simultaneous mis-operation of five separate valves contrary to established procedures; and (2) a change in process conditions that would require the operator to risk imminent death or injury. The applicant stated that in these cases no explicit administrative or engineered controls were necessary to meet the DCP. The staff responded that, in the first case, the fact that there are five valves in series is a feature of the design that must be placed under configuration control. In the second case, the threat of imminent death or injury is a deterrent only to deliberate, and not accidental, acts. The applicant stated that a configuration change involving the valves

would be reviewed by NCS personnel to ensure that it does not impact the safety basis of the facility because NCS has a permanent position on the Plant Safety Review Committee (PSRC) (USEC, 2005c). This ensures NCS review of all changes, not just changes to FMOs. The applicant clarified that the example involving threat of imminent death or injury was intended to apply to obviously life-threatening situations, such as driving a forklift into a UF_6 cylinder or cutting into a UF_6 pipe. This did not apply to accidental actions, such as violating criticality safety requirements in plant procedures. NRC agreed that, given the additional explanation, the specific examples provided were reasonable.

Section 5.4.2 of the LA (USEC, 2006b) describes how elements within the Fire Protection program, the Radiation Protection program, and the Fundamental Nuclear Material Control Plan (FNMCP) could be relied on as items related to NCS. For example, the fire protection program contains elements that can be relied on for maintaining low combustible loading or reducing the likelihood of sprinkler activation. For all items related to NCS, the applicant stated in its response to RAI NC-14 (USEC, 2005a) that "the parameters or conditions relied on and the limits must be specified in the NCSEs and controlled," meaning that specific actions or requirements within these non-NCS programs will be relied on along with the safety function they perform (e.g., maintaining operations within specified limits). Section 5.4.2 of the LA (USEC, 2006b) contains the requirements to both specify and justify the parameters and conditions relied on. There is no restriction stating that double contingency controls cannot be used as aspects of other programs, and therefore, that the applicant will identify specific aspects of items related to NCS in NCSEs and control them (through appropriate management measures) is acceptable to the staff.

With regard to RAI NC-16 regarding the NCSE approval process, the applicant stated that NCSEs require PSRC approval whenever a change impacts the ISA Summary (USEC, 2006a) (USEC, 2005a). Section 5.4.2 of the LA (USEC, 2006b) added the further clarification that PSRC approval is required for initial approval as well as changes that impact the ISA Summary (USEC, 2006a).

These requirements provide an additional layer of approval beyond approval by the NCS organization, which is appropriate when changes could affect other safety disciplines. The staff finds that the interdisciplinary nature of the PSRC provides an appropriate venue for evaluating changes that may affect the multiple disciplines that are evaluated in an integrated fashion in the ISA.

The staff determined based on the above that the applicant has presented an adequate process for establishing an adequate criticality safety basis for plant operations and performing, documenting, and approving NCSEs.

This process is consistent with expectations in NUREG-1520 (NRC, 2002). The requirements in this section of the LA (USEC, 2006b) satisfy the acceptance criteria in Section 5.4.3.4(1) and (2) of NUREG-1520 (NRC, 2002). The description of the process evaluation and approval practices provide the envelope within which NCS controlled parameters and limits are applied and determined. This process ensures that NCS limits are derived from facility NCSEs, satisfying the acceptance criteria in Section 5.4.3.4.1(10)(b) and (10)(c) of NUREG-1520 (NRC, 2002). The acceptance criteria that limits must take into consideration changes in operating parameters to ensure subcriticality, discussed in Section 5.4.3.4.1(10)(d) of NUREG-1520 (NRC, 2002), as well as variability and uncertainty, as discussed in Section 5.4.3.4.1(10)(e) and

(10)(f) of NUREG-1520 (NRC, 2002) is ensured by the applicant's compliance with the DCP (see SER Section 5.3.10). The applicant did not contain a reference to the acceptance criteria in Section 5.4.3.4.4(6) of NUREG-1520 (NRC, 2002) with regard to performing studies to correlate the change in values of controlled parameters to k_{eff}. The staff considers this unnecessary, because the applicant must demonstrate subcriticality under both normal and credible abnormal conditions, including allowing for any variability and uncertainty in parameter values. This bounding nature of facility safety limits is also ensured by the requirements to establish and maintain controlled parameters as discussed in Section 5.3.4.5.1 of this SER.

The requirements for developing NCSEs ensures that controlled parameters will be maintained during normal and credible abnormal (except for the parameter over which control is lost) conditions, satisfying the acceptance criteria in Section 5.4.3.4.2(5) of NUREG-1520 (NRC, 2002).

5.3.4.2.1 Non-Fissile Material Operations

Section 5.4.2.1 of the LA (USEC, 2006b) discusses statements regarding situations in which operations are limited to less than 1wt% ^{235}U or less than 100 g ^{235}U (referred to as "non-fissile material operations"). Both of these situations are widely known to be incapable of supporting critical configurations. ANSI/ANS-8.1-1998 (ANSI/ANS, 1998) provides a subcritical enrichment limit of 0.96wt% for homogeneous mixtures of uranium oxide and water. Although the 1wt% criteria slightly exceeds this absolute limit, the mass required for criticality approaching the 0.96wt% limit from above grows asymptotically, as indicated in the widely accepted criticality safety handbook LA-10860-MS, "Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U" (LANL, 1986), so that the minimum critical mass at 1wt% ^{235}U is extremely large. Furthermore, ANSI/ANS-8.1-1998 (ANSI/ANS, 1998) provides a subcritical mass limit of 1070 g ^{235}U for homogeneous mixtures of UO_2F_2 and water at an enrichment of 10wt% ^{235}U. A UO_2F_2-water mixture is a reasonable representation of the chemical form resulting from a moderated release of UF_6 at the ACP. This value exceeds the proposed non-FMO threshold of 100 g ^{235}U. The staff therefore finds these limits to be acceptable.

The staff noted, however, that the LA (USEC, 2006b) states: "Controls are sometimes applied to a non-fissile material operation to ensure it does not inadvertently involve fissile material. These controls...may be incorporated into applicable operating procedures or work instructions at the discretion of the responsible line manager." This appears to indicate that such controls are not always imposed and that their need is subject to line management rather than to NCS approval. In its response to RAI NC-17 (USEC, 2005a), the applicant clarified that the enrichment and mass thresholds would be applied to both normal and credible abnormal conditions. The applicant subsequently revised Section 5.4.2.1 of the LA (USEC, 2006b) to clarify that the thresholds apply to both normal and credible abnormal conditions, and that the necessary controls would be incorporated into applicable operating procedures or work instructions "when it is determined they are needed to maintain the non-fissile material operation below either 100 g ^{235}U or 1 wt. percent ^{235}U." The limit of the responsible line manager's discretion will thus be to determine whether the operation exceeds these predetermined criteria. Therefore, a non-FMO cannot exceed either the enrichment or mass threshold values under normal or credible abnormal conditions.

5.3.4.3 Design Philosophy and Review

Section 5.4.3 of the LA (USEC, 2006b) contains statements regarding the choice of controls for criticality safety. The applicant states in this section that preference will be given to the use of engineered over administrative controls. The goals of this approach are to" (1) design equipment such that criticality safety does not depend on internal or interspersed moderation, concentration, or reflection; and (2) prevent the accumulation of fissile material in inaccessible locations, through use of favorable geometry wherever practicable. The preference for the use of engineered over administrative controls is consistent with the requirement of 10 CFR 70.64(b)(1). With regard to the choice of parameters, concentration and reflection are widely recognized throughout the nuclear industry as being among the least desirable controls because of the difficulty in controlling them. It is also a common practice to analyze systems as optimally moderated. Conversely, the use of favorable geometry is widely considered one of the most reliable controls.

In addition to the preference of engineered over administrative controls, Section 5.4.3.4.2(3) of NUREG-1520 (NRC, 2002) indicates that the order of preference should be: "(a) passive engineered; (b) active engineered; (c) augmented administrative; and (d) simple administrative." In addition, Section 5.4.3.4.4(7)(a) of NUREG-1520 (NRC, 2002) indicates that two-parameter control should be preferred over one-parameter control, to minimize the potential for common-mode failure. Section 5.4.3 of the LA (USEC, 2006b) affirms this hierarchy. This approach agrees with the acceptance criteria in NUREG-1520 (NRC, 2002) and is thus acceptable to the staff.

With regard to the preference for two-parameter control, the applicant acknowledges that it is in general preferable to maintain a diverse set of controls that distributes NCS controls over many different parameters. However, it also states that this principle cannot be applied generally to ACP operations because a majority of the controls focus on maintaining moderator control over the large masses of UF_6 (such as in product withdrawal or UF_6 cylinders). In such cases, it is only necessary to have sufficient moderator to cause criticality. The staff recognizes that, based on the nature of the operation, it will not be feasible to apply two-parameter control over large portions of the facility. The staff considers the use of moderator control under these types of situations to be appropriate and in line with standard industry practice.

Section 5.4.3.4.2(3) of NUREG-1520 (NRC, 2002) states that as well as following the control preference listed above, when using a control, the choice of the type and manner of control should be justified. Section 5.4.3 of the LA (USEC, 2006b) indicates that the applicant will justify any deviations from the preferred control hierarchy. In a subsequent response to RAI NC-19 (USEC, 2005c), the applicant stated that the vast majority of ACP operations rely on passive engineered controls, including piping, cylinders, cold traps, and other equipment. Much of the process equipment is also favorable geometry. The staff concurred with this assertion based on its tours of the Lead Cascade and review of the process design during its on-site licensing review. The staff concludes that: (1) the process as designed does comply in a majority of cases with the preferred design philosophy, in that heavy reliance is placed on passive moderator or geometry control; and (2) any deviations from this in the future will be justified in plant safety basis documents. Therefore, the staff finds this to be acceptable.

5.3.4.4 Criticality Accident Alarm System Coverage

Section 5.4.4 of the LA (USEC, 2006b) contains a description of the CAAS covering ACP operations. The applicant stated in Section 1.4.1 of the LA (USEC, 2006b) that the CAAS will comply with 10 CFR 70.24 and ANSI/ANS-8.3-1997 (ANSI/ANS, 1997a), as modified by Regulatory Guide 3.71 (NRC, 2004b). This satisfies the acceptance criteria in Section 5.4.3.4.3(2) of NUREG-1520 (NRC, 2002). Section 5.4.4 of the LA (USEC, 2006b) states that some areas will not be covered by the CAAS, including UF_6 cylinder yards and areas with masses less than 700 g ^{235}U. (The cylinder yard exception is discussed in Section 5.3.6 of this SER). Non-FMOs and areas with less than 700 g ^{235}U do not require CAAS under 10 CFR 70.24(a).

The applicant stated that the CAAS is designed to detect neutron radiation resulting from the minimum accident of concern as defined in ANSI/ANS-8.3-1997 (ANSI/ANS, 1997a) and will provide an audible evacuation alarm and building radiation warning lights and alarms at both the Area Control Room (ACR) (Building X-3012) and Emergency Operations Center (Building X-1020). Every area requiring CAAS coverage will have a monitoring system consisting of at least two independent detection units, such that CAAS coverage will not be lost if one unit is out of service. 10 CFR 70.24, however, requires coverage of each area by at least two detectors. In its response to RAI NC-23 (USEC, 2005a), the applicant stated that the CAAS will be required to provide dual coverage at all times. This description of the CAAS system applies to the entire facility (except those portions that are not required to have a CAAS), and therefore satisfies the acceptance criteria in Section 5.4.3.4.3(3) of NUREG-1520 (NRC, 2002).

In the unlikely event CAAS coverage is lost, appropriate compensatory measures will be imposed. The applicant stated in Section 5.4.4 of the LA (USEC, 2006b) that it would plan and document compensatory measures as part of off-normal operation procedures before initiation of operations. Section 5.4.4 of the LA (USEC, 2006b) also indicates that these may include equipment shutdown, limiting access, or halting movement of uranium-bearing material. These compensatory measures satisfy the acceptance criteria in Section 5.4.3.4.3(7) of NUREG-1520 (NRC, 2002). These measures are consistent with usual industry practice, and are therefore acceptable to the staff. Each detection unit, moreover, will be composed of three neutron sensitive detectors, two of which will be required to initiate an alarm. This provides both for redundancy in design and elimination of spurious alarms, which can cause unnecessary risk to plant personnel from the interruption of operations and evacuation.

The applicant further stated that exterior warning lights will be provided, and facilities within 200 feet of a building with CAAS coverage will have evacuation alarms and warning lights on the exterior. This satisfies the acceptance criteria in Section 5.4.3.4.3(6) of NUREG-1520 (NRC, 2002). In its response to RAI NC-24 (USEC, 2005c), the applicant stated that the technical basis for the 200 foot radius was a neutron and gamma dose calculation performed in accordance with Regulatory Guide 3.34 (NRC, 1979). An initial radiation burst resulting from 10^{18} fissions would produce a dose of 9.166 rads at 200 feet. By comparison, ANSI/ANS-8.3-1997 (ANSI/ANS, 1997a) defines an “excessive dose” as 12 rads of combined neutron and gamma radiation. This is expected to be conservative, as 10^{18} fissions exceeds the initial radiation burst of most of the historical criticality accidents, and the model used assumes no intervening shielding. In addition, criticality alarms provide no protection against the radiation from the initial burst, but can only protect personnel from subsequent bursts. The conservative model of Regulatory Guide 3.34 (NRC, 1979) assumes 1.92×10^{17} fissions for each subsequent burst, which would result in doses of 1.76 rad for each burst. This is consistent with the current 200-foot practice at the GDPs, and is well below the definition of an “excessive dose” in the

applicable ANSI standard. Therefore, the 200-foot radius is acceptable to the staff.

In addition, the CAAS will include redundant decision logic, a backup power supply, and other diagnostic information to ensure it is reliable and available to perform its function. The backup power supply satisfies the acceptance criteria in Section 5.4.3.4.3(8)(d) of NUREG-1520 (NRC, 2002). As indicated in the applicant's response to RAI NC-25 (USEC, 2005a), the CAAS will be designed to withstand credible abnormal events, including the same abnormal events that the building structure must survive. Section 5.4.4 of the LA (USEC, 2006b) states that the CAAS must survive "credible abnormal events as described in the accident analysis for a sufficient time to warn personnel to evacuate." It is not conceivable that the CAAS will survive if the building collapses, and the collapse of the building will constitute a much graver risk to plant personnel than occurrence of an accidental criticality. Furthermore, once evacuation has been performed, the CAAS no longer provides a necessary safety function. This represents an acceptable alternative approach to the acceptance criteria in Section 5.4.3.4.3(4) and (5) of NUREG-1520 (NRC, 2002). Therefore, the staff finds these measures acceptable.

The commitments to ANSI/ANS-8.3-1997 (ANSI/ANS, 1997a) and ANSI/ANS-8.23-1997 (ANSI/ANS, 1997c) together ensure that plant personnel will be appropriately trained in responding to a CAAS alarm signal, as discussed in the acceptance criteria in Section 5.4.3.3(1)(b) of NUREG-1520 (NRC, 2002). The description of the CAAS system meets the acceptance criteria in Section 5.4.3.4.3(1) of NUREG-1520 (NRC, 2002). Although the LA (USEC, 2006b) does not specifically address personnel dosimetry, this is covered as part of the radiation protection program, and so a specific reference to the acceptance criteria in Section 5.4.3.4.3(8)(c) of NUREG-1520 (NRC, 2002), is not necessary.

Based on the statements that it would follow ANSI/ANS-8.3-1997 (ANSI/ANS, 1997a) and minimize dose and other risk to plant personnel, the staff finds these acceptable to implement the CAAS requirements of 10 CFR 70.24(a).

5.3.4.5.1 Portable CAAS

Section 5.4.4.1 of the LA (USEC, 2006b) describes the portable CAAS system (which will not meet all requirements of ANSI/ANS-8.3-1997 (ANSI/ANS, 1997a)), which may be used in the event that FMOs are performed beyond the detection range of permanently-installed CAAS detectors. The same detection capabilities will apply, although portable CAAS may be based on detection of gamma radiation. In the event a criticality accident occurs within the range of a portable CAAS, there will be an immediate audible alarm within a localized range and telemetric link to both the ACR (Building X-3012) and the Emergency Operations Center (Building X-1020). This will allow the plant's public address system to warn personnel within 200 feet of the location of the criticality. In its response to RAI NC-26 (USEC, 2005b), the applicant stated that the audible range of the portable CAAS will be 65 feet and provided dose calculations based on Regulatory Guide 3.34 (NRC, 1979) to support this (SER Section 5.3.4.5). This audibility radius of 65 feet has also been added to Section 5.4.4.1 of the LA (USEC, 2006b) and in the change pages (USEC, 2006e). Although the dose at 65 feet is calculated to exceed the 12 rad "excessive dose" defined in ANSI/ANS-8.3-1997 (ANSI/ANS, 1997a), these calculations are based on a number of conservative assumptions (e.g., large fission yield, neglect of intervening shielding). Furthermore, while the alarms will be ensured to be audible out to 65 feet, it is very probable that personnel within a larger radius will be notified in time to flee subsequent bursts. The use of a portable CAAS system will be limited to hands-on activities, such as maintenance,

that are only authorized on a temporary basis. Thus, the staff considers it unlikely that plant personnel will be exposed to more than an excessive dose as a result of not having sufficient warning to evacuate before subsequent bursts can occur. Although experimental data suggests that the time period between bursts could be much shorter than that assumed in the accident model (LANL, 2000), the fact that there is a telemetric link from the detector to the X-3012 ACR and X-1020, means there will be a very short delay between alarm actuation and notification of personnel within 200 feet via the plant public address (PA) system. Because of the 65 foot audibility radius, conservatism in the calculations, the anticipated presence of workers, and the telemetric link, the staff finds use of a portable CAAS with these requirements acceptable for short time periods. Section 5.4.4.1 of the LA (USEC, 2006b) states that the portable CAAS will not be used for more than 24 continuous hours. If this time period is exceeded, specific measures must be taken to limit exposure to personnel and the affected operations will be shut down as soon as safely achievable. This time period is sufficiently short to limit residual risk to plant personnel, and provides acceptable controls in the event that slightly longer time periods are needed. This satisfies the acceptance criteria in Section 5.4.3.4.3(7) of NUREG-1520 (NRC, 2002) with regard to time limits for compensatory measures. This is acceptable to the staff.

Together, these requirements for the permanently installed and portable CAAS ensure that an inadvertent criticality will be promptly detected to ensure that radiation exposures to workers are minimized, in accordance with the acceptance criteria in Section 5.4.3.4.1(4) of NUREG-1520 (NRC, 2002).

5.3.4.5 Technical Practices

The description of technical practices in the LA (USEC, 2006b) entails application of controlled parameters for criticality control and proper performance of criticality calculations. Together, these practices (described in the following sections) ensure that NCS determinations are performed using acceptable methodologies, satisfying the acceptance criteria in Section 5.4.3.4.1(1) of NUREG-1520 (NRC, 2002).

5.3.4.5.1 Application of Parameters

Section 5.4.5.1 of the LA (USEC, 2006b) contains requirements for the use of criticality control based on the criticality safety parameters listed below. The staff evaluated the applicability of the acceptance criteria in Section 6.4.3.4.2(7)-(16) of NUREG-1520 (NRC, 2002), to ACP operations. Many of these acceptance criteria were not applicable, primarily because of the passive nature of the ACP design.

Together, the following requirements on the various criticality control parameters ensure that NCS limits on controls and controlled parameters will be established to ensure subcriticality in accordance with the acceptance criteria in Section 5.4.3.4.1(2) of NUREG-1520 (NRC, 2002). These requirements ensure that limits are established based on credible optimum conditions, unless specific controls have been implemented on parameters to specific values, thereby satisfying the acceptance criteria in Section 5.4.3.4.1(10)(a) of NUREG-1520 (NRC, 2002). Requirements on measurement of parameters for criticality control, discussed in the acceptance criteria in Section 5.4.3.4.2(6) of NUREG-1520 (NRC, 2002), are discussed under the following sections for individual parameters.

Moderation

The applicant stated that it would assume either the optimum or worst credible moderation level when moderation is not controlled for criticality safety. Moderation will be controlled primarily by the use of passive equipment barriers (e.g., piping or cylinders). In the few cases in which it is controlled by measurement, the applicant has stated that it would use dual independent sampling. Both of these approaches are common means of implementing moderation control in nuclear industry practice.

In the revised response to RAI NC-27 (USEC, 2006c), the applicant stated that fissile material operations will not use hydrocarbon oils or uranium mixtures and solutions in ACP operations. Preferred lubricants will be polyfluoropolyethers, which are fluorinated lubricants that do not contain hydrogen (USEC, 2006d). Vacuum pumps containing hydrocarbon oil (not having direct contact with fissile material) will have a limited volume of oil that is expected to be insufficient to allow accumulation of more than 100 g ^{235}U. Although this statement is not specifically included in Section 1.1.10 of the LA (USEC, 2006b), Section 3.15 of the ISA Summary (USEC, 2006a) states that the amount of hydrocarbon-based lubricants in the few cases in which it must be used will be small enough to ensure that criticality concerns are minimal (USEC, 2006d).

In addition, the applicant's calculations demonstrate that an optimally moderated UO_2F_2-water mixture will be more reactive than a uranium tetrafluoride (UF_4)-oil system at maximum credible uranium loading. Section 5.4.5.1 of the LA (USEC, 2006b) states that "When moderation is not controlled, either optimum moderation or worst credible moderation is assumed as the normal case when performing analyses. When moderation is controlled, credible abnormal process upset conditions determine the worst-case moderated conditions." Together, these requirements provide assurance that the amount of hydrocarbon oil used will be limited, and that, regardless of whether water or oil is the moderating agent, the most reactive moderating conditions will be evaluated.

As discussed in Section C.2 of this SER, the exact location of hydrocarbon oil pumps and the amounts of lubricants to be used has not yet been determined. Section 3.1.2 of the LA (USEC, 2006b) requires the applicant to evaluate the final design against the ISA to ensure that the ISA accurately reflects the ACP design and operations, identifies all credible accident sequences, and credits the IROFS necessary to meet the performance requirements of 10 CFR 70.61. The introduction of large amounts of hydrocarbon oils would necessarily introduce new accident sequences and so would have to be reflected in the updated ISA Summary (USEC, 2006a). Therefore, based on these considerations (the limited use of small amounts of lubricant oils, the bounding nature of UO_2F_2-water systems, and the impact of the introduction of significant quantities of moderator on the accident sequences in the ISA), the staff considers the commitments regarding the use of hydrocarbon oils as described in the LA (USEC, 2006b) acceptable.

In the main processing portion of the facility, favorable geometry equipment, the design of the process, and the fluorinating environment produced by gaseous UF_6 provides for moderator control. Water is primarily available outside the process buildings, as in the UF_6 cylinder yards. The impact of moderator intrusion into UF_6 cylinders in the storage yards is discussed in Section 5.3.6 of this SER. The other major potential source of water is from firefighting activities. In the LA (USEC, 2006b), the applicant stated that, in areas containing greater than safe masses of uranium, restrictions on firefighting activities will limit the use of moderating material. In its

response to RAI NC-28 (USEC, 2005a), the applicant clarified that qualified firefighting personnel will be trained to limit use of moderating materials, but that because the impact on health and safety from a large fire would exceed that from a criticality accident, the use of moderating materials cannot be entirely prohibited. Section 5.4.5.1 of the LA (USEC, 2006b) states that pre-fire plans must contain any "unique firefighting strategy or tactics that may be needed to limit the use of moderating material." As stated in the applicant's RAI response (USEC, 2005a), firefighters must be trained on these strategies, which may include not spraying water directly into process gas openings, using fogging rather than spray nozzles where practicable, and maintaining at least 15 feet between personnel and equipment where criticality could occur. This satisfies the acceptance criteria in Section 5.4.3.4.2(12)(f) of NUREG-1520 (NRC, 2002).

The staff concurs that the chemical hazard from any released UF_6 or hydrofluoric (HF), and the hazard from a large fire itself, constitutes a greater risk to health and safety of workers and the public than a criticality accident. A criticality accident is mainly a localized event which can result in a lethal radiation dose only to those in the immediate vicinity of the reacting material. Therefore, it is reasonable that the use of moderating firefighting materials will not be entirely prohibited; the use of firefighting strategies discussed above should provide for reasonable protection against both a large-scale fire and an accidental criticality. In addition, the applicant further stated that operations containing greater than a safe mass of fissionable material will be evaluated to be subcritical even the worst credible moderation conditions resulting from fighting a large fire. Based on the above limitations and evaluation of subcriticality, the staff finds this approach acceptable for minimizing the risk to workers from an inadvertent criticality during firefighting activities.

Section 5.4.3.4.2(12)(a)–(g) of NUREG-1520 (NRC, 2002) contains acceptance criteria for the use of moderation control. The staff finds that the statements in the LA (USEC, 2006b) represents an acceptable alternative approach to these acceptance criteria for ACP operations. In the applicant's response to RAI NC-29 (USEC, 2005c), the applicant stated that a statement that it would control process variables that can affect moderation (acceptance criteria (b)) is not needed, because moderator control is provided by passive barriers as well as the nature of the process in the majority of cases, as discussed above. In the applicant's response to RAI NC-30 (USEC, 2005a), the applicant stated that compliance with ANSI/ANS-8.22-1997 (ANSI/ANS, 1997b) is not needed to provide adequate moderator control, again because of the design and nature of the process. The implementation of passive moderation control is much less complex than in situations requiring either active or administrative monitoring or control. Therefore, there is no need to address the various acceptance criteria in this section of NUREG-1520 (NRC, 2002) to provide for acceptable moderator control. The passive nature of moderator controls at the ACP satisfies acceptance criteria (d) and (g) of this section, where as acceptance criteria (a), (c), and (e) are not applicable for the reasons stated above. (See discussion on firefighting above for acceptance criteria (f)).

The major process features that ensure low risk from moderator intrusion are: (1) the use of a passive engineered (and often favorable geometry) confinement boundary; (2) a process design that limits the amount of material that could become moderated because of wet air inleakage; (3) the fluorinating UF_6 environment, which will remove some of the moderating material in the form of gaseous HF; and (4) the very limited use of moderators as part of the normal operation of the process. The commitments with regard to moderator control are therefore acceptable to the staff.

Volume

The applicant stated that it would evaluate the basis for volume control in NCSEs, and that when volume control is used, the size of containers will be controlled through the Configuration Management Program and/or by procedurally limiting the use of certain size containers. This reliance on fixed geometry for volume control satisfies the acceptance criteria in Section 5.4.3.4.2(16)(a) of NUREG-1520 (NRC, 2002). The evaluation of safe volume must include the worst-case combination of other parameters, which includes spherical geometry, optimum moderation, and full reflection. Because of these very conservative assumptions and the passive nature of these controls, this approach is acceptable to the staff.

Section 5.4.3.4.2(16)(b) of NUREG-1520 (NRC, 2002) states that whenever volume is measured, it must be done using instrumentation. For this particular facility, the staff considers this commitment to be unnecessary, and the LA (USEC, 2006b) does not contain this specific requirement. Because volume is controlled based on fixed geometry, configuration control must ensure that volume-controlled containers meet the specified volume limits. This requires measurement of the volume using some kind of physical device (instrumentation). Therefore, the staff considers the use of volume control covered by ensuring that volume control will be implemented by fixed geometry subject to configuration control.

Interaction

The applicant stated that it would evaluate the basis for spacing requirements in NCSEs and to use passive engineered controls to the extent possible to ensure spacing is maintained. In these cases, the structural integrity must be sufficient to maintain spacing under normal and credible abnormal conditions. Because of to the passive nature of most of these controls, this approach is acceptable to the staff. This discussion of the physical integrity satisfies the acceptance criteria in Section 5.4.3.4.2(14)(a) of NUREG-1520 (NRC, 2002).

Geometry

The applicant stated that it would specify geometry controls in NCSEs, maintain them as part of the Configuration Management Program, and verify them before initiating operations. The basis for geometry controls may be taken from established standards or calculations. In its response to RAI NC-31 (USEC, 2005a), the applicant stated when safe parameter limits are taken from handbooks, they will be supported by calculations “to ensure credible abnormal events are adequately subcritical.” The applicant clarified in a subsequent call that the intent of this is not that all values taken from handbooks be supported by calculations. Normal condition values may be taken from suitable handbooks (which are discussed in Section 5.3.4.5.2 of this SER), but calculations may be necessary to demonstrate that the abnormal conditions that must then be evaluated are subcritical. Margins to ensure subcriticality for handbook-derived values are also discussed in the response to RAI NC-44 (Section 5.3.4.6.2 of this SER).

Section 5.4.5.1 of the LA (USEC, 2006b) states that geometry controls will be maintained by the facility’s configuration control program, and Section 5.4.3 of the LA (USEC, 2006b) states that credit may be taken for nuclear and physical characteristics of equipment and materials if control is exercised to maintain them if they can credibly degrade. During the on-site licensing reviews, the staff observed that the majority of ACP operations relying on geometry control are based on the passive design of equipment, much of which is composed of robust structures with

a high degree of structural integrity (such as UF_6 cylinders). The environmental conditions to which this equipment is subjected are not expected to result in significant degradation. Therefore, the staff has reasonable assurance that geometry controls will be maintained over the lifetime of the facility.

The applicant's commitment to use configuration control satisfies the acceptance criteria in Section 5.4.3.4.2(8)(a) of NUREG-1520 (NRC, 2002).

Mass

The applicant stated that it would determine safe mass values in NCSEs. Safe mass values will be based on established standards or calculations. The applicant stated that when mass is not controlled, the greatest credible mass will be assumed based on available volume, and that instrumentation will be used when mass is based on measurement. The commitment to use instrumentation satisfies the acceptance criteria in Section 5.4.3.4.2(7)(c) of NUREG-1520 (NRC, 2002).

The applicant also stated that the safe mass values depend on many factors, including the geometry, enrichment, and composition. However, Section 5.4.5.1 of the LA (USEC, 2006b) also states that the combination of mass and these other dependent factors will only be used as one leg of double contingency. This therefore satisfies the acceptance criteria in Section 5.4.3.4.2(1) of NUREG-1520 (NRC, 2002), and will ensure that the two legs of double contingency are independent.

The meaning of the term "safe mass" is defined in Section 5.4.5.1 of the LA (USEC, 2006b) as no more than 43.5% of the minimum critical mass (USEC, 2006c). Section 5.4.3.4.2(7)(d)-(e) of NUREG-1520 (NRC, 2002) states that when double batching is credible, masses should be limited to less than 45% of the minimum critical mass, and that when double batching is not credible, masses should be limited to less than 75% of the minimum critical mass. Since no batch operations will be performed at the ACP, and since double batching is not credible, the staff finds that limiting safe mass values to less than half the minimum critical mass is acceptable.

The commitment that the "greatest credible mass will be assumed based on available volume" ensures that the acceptance criteria in Section 5.4.3.4.2(7)(a) and (7)(b) of NUREG-1520 (NRC, 2002) to consider the mass percentage of special nuclear material and conservative process densities will be satisfied. If a limiting mass percentage and process density are not assumed, then the applicant cannot ensure that the greatest credible mass based on the available volume has been determined.

Enrichment

The applicant stated that it would limit ACP operations to less than 10wt% ^{235}U, except for small quantities of material in the form of laboratory samples or standards. Because the real-time, accurate measurement of enrichment is difficult, the maximum credible enrichment (10wt% ^{235}U) will be assumed in calculations for all operations, while actual material to be produced will not initially exceed 5wt% ^{235}U. As discussed in Section 1.2.3.4 of this SER, the applicant has also stated that it will notify NRC 60 days before increasing the enrichment above 5wt% ^{235}U. (Assuming 10wt% ^{235}U while limiting operations to 5wt% ^{235}U provides a large margin of safety.

Above 5wt% ^{235}U, the decreased conservatism will be offset by the increase in the margin of subcriticality from 0.02 to 0.05.) In the few instances in which enrichment control is relied on for criticality safety, the measurement will be obtained by installed equipment or laboratory sample. This satisfies the acceptance criteria in Section 5.4.3.4.2(10)(b) of NUREG-1520 (NRC, 2002). In the applicant's response to NRC's RAI NC-35 (USEC, 2005a), the applicant stated that the only operations that rely on enrichment controls are for UF_6 cylinders, which have to meet ANSI specified enrichment limits for off-site transportation (though not for on-site storage). While these cylinders must be limited to 5wt% ^{235}U in order to be used for transportation, the applicant stated that cylinders will be shown to be subcritical for enrichments up to 10wt% ^{235}U. Exceeding the allowed enrichment in a UF_6 cylinder would first require a massive failure in the enrichment cascade. The staff's review of measures to prevent exceeding the maximum authorized and analyzed enrichment of 10wt% ^{235}U is described in Section C.1.1 of this SER. The staff finds use of the maximum site-wide enrichment of 10wt% ^{235}U, except where controlled by measurement, to be acceptable.

Because the maximum on-site enrichment is assumed in criticality calculations (except as noted above), a method of segregating materials of different enrichments is not needed, as discussed in the acceptance criteria in Section 5.4.3.4.2(10)(a) of NUREG-1520 (NRC, 2002).

Density

The applicant stated that it would justify any cases of reliance on density control in NCSEs, and in such cases, to measure density using instrumentation, in accordance with the acceptance criteria in Section 5.4.3.4.2(9)(b) of NUREG-1520 (NRC, 2002). The majority of processes use either gaseous or solid UF_6. Gaseous UF_6 has very low density; systems involving it will have very low reactivity. Solid UF_6 will exist primarily in cylinders and is not readily controllable, so that full theoretical density is assumed. Based on its review of ACP processes, the staff is not aware of any operations relying on density control. However, if density is controlled, the above statements are acceptable to the staff.

Section 5.4.3.4.2 of NUREG-1520 (NRC, 2002) states that process variables that can affect density should be identified as IROFS in the ISA Summary (USEC, 2006a). If density is a controlled parameter and is credited in the ISA Summary (USEC, 2006a), then appropriate controls should be identified as IROFS in accordance with the applicant's ISA methodology (see Chapter 3 of this SER). If density is not controlled or credited in the ISA Summary (USEC, 2006a), then such a commitment is unnecessary.

Heterogeneity

The applicant stated that homogeneous safe mass values at 10wt% ^{235}U are sufficient to bound heterogeneous systems at 10wt%, because the homogeneous and heterogeneous minimum critical mass values are very close. While there may be small differences between the limits associated with homogeneous and heterogeneous systems at low enrichments, the staff is not aware of any operations that would lead to heterogeneous configurations. Fuel assemblies are not manufactured at the ACP, and there are no operations involving mixing of solid uranium with moderator (such as in a blender). Therefore, a specific commitment to the acceptance criteria in Section 5.4.3.4.2(4) of NUREG-1520 (NRC, 2002), is unnecessary.

Concentration

The applicant stated that concentration control is not applied to any operations involving more than a safe mass of material, but that in cases where concentration control is relied on, dual independent sampling will be used. In its responses to RAIs NC-37 and NC-38 (USEC, 2005a), the applicant further clarified that there are no FMOs relying on concentration control. Because of these statements regarding dual sampling and the lack of reliance on concentration control, the staff finds this to be acceptable.

Because the applicant does not rely on concentration control, the acceptance criteria in Section 5.4.3.4.2(13)(a)–(c) and (e) of NUREG-1520 (NRC, 2002) do not apply. Acceptance criteria 5.4.3.4.2(13)(d), regarding dual, independent sampling, is met, as discussed above.

Reflection

The applicant stated that it would consider normal and credible abnormal conditions, including the possibility of full water reflection, in NCSEs. The possibility that concrete can be a more efficient reflector than water will also be considered. Section 5.4.5 of the LA (USEC, 2006b) states that, in floor-level operations involving the routine presence of personnel, either full water reflection, or interstitial moderation combined with full density blocks of water, would be used to bound the presence of personnel. For elevated equipment around which the routine presence of personnel is not credible, full water reflection would not be considered. Although it did not specify a minimum reflection condition, the applicant did state that it would include bounding reflection conditions to cover normal and credible abnormal conditions. Given the lack of liquids or solution processing in the ACP, the staff finds this to be acceptable.

In the applicant's response to RAI NC-40 (USEC, 2005a), the applicant stated that full flooding is not credible, given the height of the ACP above the flood plain and the lack of rooms or confined areas that could allow such an accumulation of water. Section 5.4.5.1 of the LA (USEC, 2006b) states that, to cover sprinkler activation, the applicant will consider a thin film of water and a low-density mist between components. Based on its review of the operation, the staff concurs that occurrence of full flooding is very unlikely. Based on its knowledge of interstitial moderation studies for equipment arrays at other facilities, the staff has reasonable assurance that assuming a thin film and low density mist will bound the interstitial moderation resulting from sprinkler activation. The above statements with regarding bounding reflection satisfy the intent of the acceptance criteria in Section 5.4.3.4.2(11)(a) of NUREG-1520 (NRC, 2002) which is to ensure that bounding reflection conditions are assumed in all calculations.

Section 5.4.3.4.2 of NUREG-1520 (NRC, 2002) states that controls to prevent the presence of reflectors should be identified as IROFS in the ISA Summary (USEC, 2006a). If reflection is a controlled parameter and is credited in the ISA Summary (USEC, 2006a), then appropriate controls should be identified as IROFS in accordance with the applicant's ISA methodology (see Chapter 3 of this SER). If reflection is not controlled or credited in the ISA Summary (USEC, 2006a), then such a commitment is unnecessary.

Neutron Absorption

The applicant stated that it would comply with the requirements of ANSI/ANS-8.21-1995 (ANSI/ANS, 1995) when relying on neutron absorber control. These requirements include sampling and periodic inspection to ensure the neutron absorbers remain effective.

Uncertainties because of manufacturing tolerances, corrosion, chemical reactions, neutron spectra, and neutron cross sections will be considered. In its response to RAI NC-41 (USEC, 2005a), the applicant stated that it would follow ANSI/ANS-8.21-1995 (ANSI/ANS, 1995), but stated that it currently did not have any FMOs relying on fixed neutron absorbers. In its response to RAI NC-42 (USEC, 2005a), the applicant stated that it did not rely on the use of borosilicate glass raschig rings, and thus a statement that it will follow ANSI/ANS-8.5-1996 (ANSI/ANS, 1996a) is not necessary (as discussed in the acceptance criteria in Section 5.4.3.4.2(15)(a) of NUREG-1520 (NRC, 2002)). The staff therefore finds this to be acceptable.

Section 5.4.3.4.2(15)(c) of NUREG-1520 (NRC, 2002) states that neutron spectra should be evaluated when evaluating absorber effectiveness. Although the LA (USEC, 2006b) does not contain a specific requirement to this effect, the staff considers this to be unnecessary. In practice, calculations would be performed whenever credit is taken for neutron absorbers, and the nature of the calculational methods discussed in Section 5.3.4.5.2 of this SER is such that it must consider the neutron flux and energy spectrum to determine a value of k_{eff}. Thus, the intent of the acceptance criteria will be met by virtue of the codes being used.

5.3.4.5.2 Methods of Calculation

Section 5.4.5.2 of the LA (USEC, 2006b) describes a variety of methods that may be used to demonstrate subcriticality, as discussed below. These include the use of experimental data, handbooks, hand calculations, and (deterministic or probabilistic) computer calculations. The following discussion provides details on the review of the applicability, limitations of use, and validation of these methods.

Together, the following requirements on calculations and other methods of evaluation ensure that methods will be validated, appropriate assumptions will be used, and acceptable computer codes (when used) will be employed, in accordance with the acceptance criteria in Section 5.4.3.4.1(3) of NUREG-1520 (NRC, 2002).

Experimental Data

Experimental data is used to validate computer calculations (see Appendix C of this SER), but not to evaluate operations. In its response to RAI NC-43 (USEC, 2005a), the applicant clarified that its statement to the effect that “the generic nature of the experimental data does not address the variables present in different operations” addressed the use of experimental data by themselves showing subcriticality, and did not refer to benchmark experiments used in the validation. Experimental data is not used by themselves to show subcriticality, and therefore explicit requirements in the LA (USEC, 2006b) are not needed.

Handbooks

The applicant stated that handbooks are used to evaluate certain simple systems that do not require explicit calculation. Two examples of handbooks are ARH-600, “Criticality Handbook,” (Carter et. al, 1969) and LA-10860-MS, “Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U” (LANL, 1986). These are both widely recognized and accepted in the nuclear criticality community. The LA (USEC, 2006b) does not limit the applicant to use just these handbooks. In its response to RAI NC-44 (USEC, 2005c), the applicant stated that handbooks used for ACP operations are those that are nationally recognized throughout the industry as

high quality analyses that have been confirmed through many years of use or are based on experimental data. Handbooks other than these two will be held to similar criteria for excellence, industry acceptance, and quality of data. These criteria for quality of handbooks used for criticality determinations have been included in Section 5.4.5.2 of the LA (USEC, 2006b).

Section 5.4.5.2 of the LA (USEC, 2006b) states that limits from handbooks will be reduced from the minimum critical values contained in the handbooks, so that operations can be shown to be subcritical. Although the LA (USEC, 2006b) does not describe specific margins of safety, as discussed in the acceptance criteria in Section 5.4.3.4.2(8)(b) of NUREG-1520 (NRC, 2002), the staff has reasonable assurance that the reduction of handbook limits will ensure adequate margin of subcriticality in the few instances in which handbook-derived values are used, and so this is acceptable to the staff. In addition, handbooks are not widely used in the industry for setting limits because of the increased power and availability of computer methods, but are often used in "scoping calculations" (i.e., as the starting point for calculational studies). The staff verified that handbooks were not widely used to set NCS limits for ACP operations, by examination of plant criticality calculations during the on-site review.

Hand Calculations

The applicant stated that the following hand calculations may be used to show subcriticality: the modified two-group diffusion equation, buckling conversion, and comparative analysis for evaluating single units; and the solid angle and surface density method for evaluating arrays. The two-group diffusion equation is applicable to simple fuel-moderator (primarily solution) systems with spherical, cylindrical, or cuboid shapes, and consists of an analytical solution to the diffusion equation with two neutron energy groups. The buckling conversion method is applicable to all types of materials, and consists of a mathematical transformation between simple geometric shapes. Comparative analysis consists of comparison with handbook data. The solid angle method is applicable to arrays of simple individual units containing solutions and is based on neutron interaction between array elements, in terms of the subtended solid angle. The surface density method is applicable to large arrays with units containing solutions or metals and is based on the areal density of units in the array. For both the solid angle and surface density method, single unit reactivity is determined using other methods.

In its response to RAI NC-45 (USEC, 2005a), the applicant stated that: (1) validation of the two-group diffusion method is not necessary because it uses experimentally-derived data, and there is inherent conservatism in the method; (2) validation of buckling conversion is not needed because it is a universal method applicable to all material compositions (but simple shapes); (3) data derived from handbooks are treated as in the previous section; (4) validation of the solid angle method is not necessary because of its inherent conservatism; and (5) validation of the surface density method is not necessary because it merely applies the reactivity of a single unit, which is determined from handbooks or validated methods, to an entire array. This response also indicated that, although there are numerous such methods available, the methods used will be those described in "Nuclear Criticality Safety" (ANS, 1991) and will be subject to an upper safety limit of 0.95. These methods, as described in this reference, are widely accepted in the nuclear industry; conformance with the methodology in the reference and the limitations described above provides reasonable assurance that they will be used correctly. In addition, the staff finds, in conjunction with the amount of conservatism inherent in these methods, that the use of a 0.95 limit will be sufficient to provide an adequate margin of subcriticality for safety.

Computer Calculations

The applicant stated that it would use critical benchmark experiments to validate computer codes, ideally using geometries and materials similar to the systems being modeled. Section 5.4.5.2 of the LA (USEC, 2006b) states that if sufficient benchmarks with similar geometries and materials are not available, one of several options will be employed. The analyst can: (1) omit the material from the model (if it can be done conservatively); (2) substitute a conservative material within the validated area of applicability (AOA); (3) demonstrate that inclusion of a specific material will not have an impact on the bias, by considering the material cross sections; (4) adjust the material density to compensate; (5) reduce the upper safety limit to account for additional uncertainty; or (6) add benchmark experiments to cover the specific material. Provided one of these methods is used, this approach will ensure that only validated materials are used or additional margin is employed, and is therefore acceptable to the staff.

In addition to determining an acceptable AOA based on benchmark experiments, the applicant stated that it would determine the bias and uncertainty in the bias, which includes uncertainty from both the calculations and experimental data, using statistical methods. The applicant will use statistical factors to ensure that there is a 95% confidence that 99.9% of future k_{eff} values below the upper safety limit are subcritical. Furthermore, the applicant stated in the LA (USEC, 2006b) that it would use the validation methods in NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (NRC, 2001b). The staff reviewed the description of the applicant's validation methodology in Section 5.4.5.2 of the LA (USEC, 2006b) and determined that it included all the criteria in Section 5.4.3.4.1(7) of NUREG-1520 (NRC, 2002) with the exception of describing the code's AOA Section 5.4.3.4.1(7)(b). The validation report (USEC, 2005d) was referred to in the LA (USEC, 2006b) as Reference 11 of Chapter 5. The LA (USEC, 2006b) stated that the applicant relied on this validation report in its calculations to support the initial licensing effort, but did not state that all future calculations would be done under this specific validation report. In meetings and telephone calls, the applicant stated that relying on a single specific validation report would be unnecessarily restrictive and would limit its ability to use other valid methods.

The applicant submitted and the staff reviewed one validation report (USEC, 2005d) in support of the applicant's requested margin of subcriticality for the ACP. Because of this, NRC's assurance of subcriticality depends on future ACP calculations being done within the envelope defined by WSMS-CRT-03-0093, Rev. 2 (USEC, 2005e). The staff recognizes that there are many insignificant changes that could be made to the validation report (e.g., editorial changes) that would not impact the technical basis for the margin of subcriticality, and therefore agrees that it is not necessary to restrict the applicant from making any changes to its validation report or from developing new validation reports. However, if the technical basis for the subcritical margin changes, then a re-review of the acceptability of the validation and the basis for the subcritical margin may be needed. The staff determined that the most significant aspects of the validation, as outlined in the acceptance criteria in Section 5.4.3.4.1(7) of NUREG-1520 (NRC, 2002) were adequately described in the LA (USEC, 2006b). Section 5.4.5.2 of the LA (USEC, 2006b) describes the validation methodology, including the statistical confidence that must be met, the minimum margin of subcriticality, and the bounds of the validated AOA. Although the description of the AOA in the LA (USEC, 2006b) is fairly detailed, it does not include all of the limitations and caveats discussed in the validation report (e.g., limiting the use of some materials (lead, concrete) to the thermal energy range). Section 5.4.5.2 of the LA (USEC, 2006b) also states that prior NRC approval will be obtained before extending the bounds of the

validated AOA as determined in the validation report. The staff finds that this provides reasonable assurance that the validated AOA will not be extended without appropriate regulatory review.

The applicant also stated that it would establish controls and limits to ensure that the maximum k_{eff} does not exceed the upper safety limit as specified in the applicable code validation. Scoping and analysis calculations are allowed to be performed using unvalidated codes, but calculations used as the basis for NCSEs must be confirmed by or performed using configuration-controlled codes validated with at least the same degree of conservatism as in validation report WSMS-CRT-03-0093, Rev. 2. In Section 5.4.5.2 of the LA (USEC, 2006b), the applicant stated that any calculations performed on an unvalidated platform must be repeated on a validated platform to be used as the basis for any NCS limit, double contingency control, or IROFS. In addition, changes to the hardware and software configuration of the system will be evaluated in accordance with 10 CFR 70.72. The staff has determined that these programmatic requirements are consistent with the intent of ANSI/ANS-8.1-1998 (ANSI/ANS, 1998) and Section 5.4.3.4.1 of NUREG-1520 (NRC, 2002), and are therefore acceptable to the staff.

5.3.5 Use of National Consensus (ANSI/ANS-8 Series) Standards

Section 1.4 of the LA (USEC, 2006b) contains references to various industry codes and standards, including certain ANSI/ANS-8 Series standards applicable to NCS. Many of these standards have been endorsed in Regulatory Guide 3.71 (NRC, 2004b), which are also listed in the acceptance criteria of Section 5.4.2 of NUREG-1520 (NRC, 2002). Section 5.4 of NUREG-1520 (NRC, 2002) states that the applicant should identify and justify any variation from the standards in the LA (USEC, 2006b).

In Section 1.4 of the LA (USEC, 2006b), the applicant has stated that it would comply with the following standards for NCS, with some exceptions. The staff's review of this discussion is discussed below:

- ANSI/ANS-8.1-1998 "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" (ANSI/ANS, 1998). This standard was endorsed in Regulatory Guide 3.71 (NRC, 2005). The use of this version is therefore acceptable. The applicant did take exception to Section 4.1.6 of the standard. Section 4.1.6 of the standard states that operations shall be reviewed annually by individuals who are knowledgeable in NCS, in consultation with operations. The applicant stated, however, that operations personnel will perform the review, and will be assisted by individuals knowledgeable in NCS. Although this is technically an exception to the standard, the staff notes that individuals from both organizations will be involved in either case. The only difference is that the effort will be headed by operations rather than NCS personnel. Given the low reliance placed on administrative controls, the staff finds this acceptable. This satisfies the acceptance criteria with regard to organization and administration in Section 5.4.3.2(2) of NUREG-1520 (NRC, 2002). This also satisfies the acceptance criteria with regard to the intent of Section 4.1.1 of the standard in Section 5.4.3.2(3) of NUREG-1520 (NRC, 2002). This also satisfies the acceptance criteria with regard to evaluation methodologies in Section 5.4.3.4.1(5) of NUREG-1520 (NRC, 2002). This also satisfies the acceptance criteria with regard to the policy against limits being accidentally exceeded in Section 5.4.3.4.4(2) of NUREG-1520 (NRC, 2002).

- ANSI/ANS-8.3-1997 “Criticality Accident Alarm System” (ANSI/ANS, 1997a). The applicant stated that it took exception to Section 1.2.5 of the standard, which does not exist. However, the substance of the exception is that areas with greater than 700 g ^{235}U will have CAAS coverage, with the exception of the UF_6 cylinder storage yards. Acceptability of this exemption from 10 CFR 70.24(a) requirements is discussed in Section 5.3.6 of this SER.

- ANSI/ANS-8.19-1996, “Administrative Practices for Nuclear Criticality Safety” (ANSI/ANS, 1996b). The applicant took exception to Section 7.8 of the standard. This is identical to the issue concerning annual review of operations, discussed above under ANSI/ANS-8.1-1998 (ANSI/ANS, 1998). For the reasons noted above, this is acceptable to the staff. This satisfies the acceptance criteria with regard to organization and administration in Section 5.4.3.2(2) of NUREG-1520 (NRC, 2002). This satisfies the acceptance criteria with regard to procedures in Section 5.4.3.3(2)(a) of NUREG-1520 (NRC, 2002). (NOTE: The other part of Section 5.4.3.3(2)(a) is covered in the commitment to the DCP discussed in SER Section 5.3.10.) This satisfies the acceptance criteria with regard to audits and assessments in Section 5.4.3.3(3)(a) of NUREG-1520 (NRC, 2002).

- ANSI/ANS-8.20-1991, “Nuclear Criticality Safety Training” (ANSI/ANS, 1991). The applicant stated that it would follow this standard without exception, which is therefore acceptable to the staff. This satisfies the acceptance criteria in Section 5.4.3.3(1)(a) of NUREG-1520 (NRC, 2002).

- ANSI/ANS-8.21-1995, “Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors” (ANSI/ANS, 1995b). The applicant stated that it would follow this standard without exception, which is therefore acceptable to the staff. This satisfies the acceptance criteria in Section 5.4.3.4.2(15)(b) of NUREG-1520 (NRC, 2002).

- ANSI/ANS-8.23-1997, “Nuclear Criticality Accident Emergency Planning and Response” (ANSI/ANS, 1997c). The applicant stated that it will follow the standard, which satisfies the acceptance criteria with regard to emergency procedures and training in Section 5.4.3.1(7) of NUREG-1520 (NRC, 2002), and the acceptance criteria with regard to emergency management in Section 5.4.3.4.3(8)(a) of NUREG-1520 (NRC, 2002).

The applicant did not rely on the other NCS standards listed in Section 5.4.2 of NUREG-1520 (NRC, 2002). The staff found this acceptable as follows:

- ANSI/ANS-8.5-1996, “Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material” (ANSI/ANS, 1996a) The applicant stated in its response to RAI NC-42 (USEC, 2005a) that ACP operations will not rely on the use of raschig rings, but rather will rely on the use of fixed absorbers for neutron absorber control, in accordance with ANSI/ANS-8.21-1995 (ANSI/ANS, 1995). Therefore, a reference to this standard is not needed.

- ANSI/ANS-8.6-1983, “Safety in Conducting Subcritical Neutron-Multiplication Measurements in Situ” (ANSI/ANS, 1983a). As a fuel enrichment facility, the ACP will not perform subcritical neutron multiplication measurements, and therefore a reference to this standard is not needed.

- ANSI/ANS-8.7-1975, “Guide for Nuclear Criticality in the Storage of Fissile Materials” (ANSI/ANS, 1975). This standard is only applicable to applications involving uranium enriched to greater than 30wt% ^{235}U, and is therefore not applicable to ACP operations. Therefore, a reference to this standard is not needed.

- ANSI/ANS-8.9-1987, “Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials” (ANSI/ANS, 1987a). The applicant has stated that it will rely on validated calculational methods or handbooks for determining NCS limits. The applicant would therefore have to evaluate any steel pipe intersections as in other processes, using validated methods as specified in ANSI/ANS-8.1-1998 (ANSI/ANS, 1998). Moreover, large volumes of solutions will not be processed at the ACP. Therefore, a reference to this standard is not needed.

- ANSI/ANS-8.10-1983, “Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement” (ANSI/ANS, 1983b). The nature of processes and materials at the ACP is such that there is no need for reliance on shielding and confinement that could be credited with reducing the consequences of a criticality accident. Criticality control will be by prevention rather than mitigation, in accordance with the DCP. Therefore, a reference to this standard is not needed.

- ANSI/ANS-8.12-1987, “Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors” (ANSI/ANS, 1987b). The ACP will not be licensed to handle critical mass quantities of plutonium, and therefore a reference to this standard is not needed.

- ANSI/ANS-8.15-1981, “Nuclear Criticality Control of Special Actinide Elements” (ANSI/ANS, 1981). The ACP will not be licensed to handle critical mass quantities of fissionable actinide elements, and therefore a reference to this standard is not needed.

- ANSI/ANS-8.17-1984, “Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR [Light Water Reactor] Fuel Outside Reactors” (ANSI/ANS, 1984). As a fuel enrichment facility, the ACP will not transport finished reactor fuel. Material received by or shipped from the ACP will consist of solid UF_6 in approved cylinders, to which this ANSI/ANS-8.17-1984 (ANSI/ANS, 1984) does not apply. Therefore, a reference to this standard is not needed.

- ANSI/ANS-8.22-1997, “Nuclear Criticality Safety Based on Limiting and Controlling Moderators” (ANSI/ANS, 1997b). The applicant stated in its response to RAI NC-30 (USEC, 2005a) that moderator control will be based mainly on passive equipment barriers (e.g., piping, UF_6 cylinders) and will not require extensive programmatic controls. This approach is acceptable to the staff, and therefore a reference to this standard is not needed.

The above review means that a specific reference to these standards, as discussed in Section 5.4.3.4.4(3) of NUREG-1520 (NRC, 2002), is not necessary.

5.3.6 CAAS Exemption

Section 1.2.5 of the LA (USEC, 2006b) requested an exemption from the criticality monitoring requirements of 10 CFR 70.24 for the UF_6 cylinder storage yards. 10 CFR 70.24(a) requires installation of a criticality detection and alarm system in each area in which greater than 700 g ^{235}U is handled, used, or stored. The basis for the applicant's request was that the frequency of a criticality accident in the cylinder storage yards is at most 5×10^{-6}/yr, that increased vehicular and pedestrian traffic would increase the likelihood of impacts involving cylinders, and that risk to workers from radiation exposure would increase because of maintenance and calibration efforts. 10 CFR 70.17 allows exemption to be granted when such relief is authorized by law, does not endanger life, nor property, nor the common defense and security, and is otherwise in the public interest.

As noted in Section 1.2.5 of the LA (USEC, 2006b), this request is similar to that previously granted for the cylinder storage yards in the GDPs (NRC, 2001a). That approval was based on: (1) integrity of the cylinders and vehicle handling practices, which make the occurrence of a large cylinder breach very unlikely; and (2) a response time that ensures that the cylinders will not accumulate sufficient moderator to make criticality possible. The cylinders and their contents at the ACP are identical to those in the storage yards at the GDPs, with the exception that material with enrichments up to 10wt% ^{235}U could be present.

Cylinders with material enriched up to 5wt% ^{235}U would be identical to those in the exempted cylinder storage yards at the GDPs. The applicant performed an analysis to show that a 10-ton cylinder filled with 5wt% ^{235}U enriched UF_6 would be subcritical even if the entire void space in the cylinder (approximately 40% of the total volume) were filled with water. Based on available meteorological data for this geographic region, accumulation of sufficient rainwater to make criticality possible would require at least several months with even a very large diameter hole in the cylinder shell (discussed in more detail below).

Staff had reviewed the applicant document POEF-2086/ORNL/TM-11988, "Investigation of Breached Depleted UF_6 Cylinders" (ORNL, 1991), during the on-site ISA review. This documented the result of a three-site survey (Portsmouth, Paducah, and Oak Ridge's K-25 GDPs) and stated that only two cylinder integrity failures had been discovered. These failures consisted of holes in two 14-ton depleted uranium cylinders, and in both cases, the hole was self-sealing such that there was only a small release. Both of these failures appeared to be caused by impacts from the lifting lugs of other nearby cylinders, rather than corrosion *in situ*. Other failures had been noted to occur during cylinder movement, but these were promptly detected and corrected. The only other historically noted cylinder failures occurred in 1978 when a cylinder was dropped onto its cradle, and in 1982 when a cylinder failed on a railcart from unknown causes. However, these latter failures occurred before changes in the sulfur content of the cylinder materials of construction so as to improve cylinder integrity. The applicant stated that the total population of cylinders in which the two failed cylinders were discovered probably exceeded 40,000. The applicant noted that the cylinders used for depleted uranium storage are some of the older cylinders in existence, that newer cylinders would typically be used for storage of enriched product, and that these cylinders would be shipped to other sites and not stored on-site for several years. Based on the above, the staff concludes that failure of an enriched product cylinder from other than a transportation mishap is extremely unlikely. Even most transportation mishaps are unlikely to result in a breach.

The staff reviewed the internal report K-D-1987, "Water Immersion Tests of UF_6 Cylinders with Simulated Damage" (ORNL, 1967), in which metal cylinders and glass vessels containing UF_6

were immersed in water under a variety of different conditions. The results of these experiments showed that, for small breaches adjacent to solid UF_6, chemical reactions between the UF_6, water, and metal caused the formation of an impervious self-protecting layer (predominantly UO_2F_2) that ensured very little water intrusion. However, when more than a small breach occurred adjacent to the void space in a cylinder, significant water intrusion was observed. Breaches occurring adjacent to the void space at the top of a solid UF_6 cylinder would also be those oriented in such a way as to allow rainwater to enter the cylinder. Therefore, it was necessary to evaluate the amount of time that a cylinder would be vulnerable to this scenario and how long it would take to accumulate a sufficient quantity of rainwater to make criticality possible.

Section 1.2.5 of the LA (USEC, 2006b) states that cylinders will be covered within 5 hours of discovery of any breach. A breach is most likely to occur as the result of an accident during cylinder movement, and thus would be detected very shortly after it occurred. Therefore, based on the amount of water that would be required to permit criticality (more than that required to fill the void space), the historically demonstrated unlikelihood of having a breach of sufficient size to allow accumulation of this amount of water, the long time frames required to accumulate this amount of water, and the statement that The applicant would promptly repair breaches upon detection, the staff has reasonable assurance that occurrence of criticality in a cylinder with material enriched up to 5wt% ^{235}U is extremely unlikely. In addition, this has the same technical basis as the previous approval.

Cylinders with between 5 and 10wt% ^{235}U would require less water to make criticality possible, and therefore this requires a different technical basis than the previous approval. The applicant determined that at least 80 liters of water would be needed to cause criticality in a 10-ton cylinder at 10wt% ^{235}U. The analysis supporting this figure recognizes that some of the water will be removed in the reaction with UF_6, and assumes that the UO_2F_2 solution will form a slab layer on top of the UF_6. The applicant estimated that at 10wt% ^{235}U, the safe slab depth would be 3.2 inches, which would take up a volume of approximately 100 liters on top of the UF_6. Based on water taking up 75% of the total layer volume and ~5 liters being consumed in the reaction with UF_6, the applicant determined that it would take 80 liters of water before criticality is possible. Staff performed an independent analysis calculating how much of the volume of a 2.5-ton cylinder (30-inch diameter by 7-feet long) would be taken up with 2.5 tons of UF_6, and by obtaining the safe slab depth of approximately 8.3 cm (3.27 inches) from Table 6 of ANSI/ANS-8.1-1998 (ANSI/ANS, 1998).

To determine the amount of time required to accumulate this amount of water, the applicant used 24-hour rainfall data and assumed the breach to consist of a 6- or 12-inch diameter hole. Table 1.3-2 of the LA (USEC, 2006b) includes the following data:

Table 5-1
Rainfall Data

Return Period	24-hour rainfall
10 years	4.01 inches
100 years	6.5 inches
1,000 years	8.36 inches
10,000 years	10.44 inches

The applicant determined that, with a 6-inch diameter hole, it would require 172 inches of rain to accumulate 80 liters of water in the cylinder. This is the volume of a right circular cylinder with a length of 172 inches and a base 6 inches in diameter, and assumes the rain is incident normal to a surface subtended by the breach. Based on this analysis, it would require 16 days of a sustained 10,000-year rainfall to accumulate 172 inches. With a 12-inch diameter hole, it would require 4 days of sustained 10,000-year rainfall for a total of 43 inches. Such large breaches would require a violent high-speed impact, since cylinders used for off-site transport must meet strict hypothetical accident tests under 10 CFR Part 71 and must be certified for transportation. Smaller breaches would eventually result in partial or complete plugging of the hole by reaction products, as has been experimentally demonstrated. Historical data shows no record of ever experiencing the type of massive breach that would be required for such an accident, as stated above. The applicant's response to RAI CA-3 (USEC, 2005a), however, indicated that cylinders with greater than 5wt% ^{235}U (in which criticality could occur with sufficient water) will not be stored outdoors. The response to RAI CA-5 (USEC, 2005a) further states that such cylinders will be stored inside existing facilities provided with CAAS coverage. (Note that these cylinders are only approved for transportation up to 5wt% ^{235}U, so that cylinders with greater than 5wt% ^{235}U will only be used for indoor storage.) However, LA (USEC, 2006b) Section 1.2.5 states that UF_6 cylinders in the cylinder storage yards are not covered by a CAAS system unless the cylinders contain UF_6 enriched to more than 5wt% ^{235}U. During a subsequent phone call, The applicant explained that the intent is for the license to prohibit storage of UF_6 cylinders enriched to more than 5wt% ^{235}U without CAAS coverage, and the RAI responses were based on its intent not to install a CAAS system in the cylinder storage yards.

Given the unlikelihood of having a large cylinder breach, because of the stringent requirements on cylinder integrity and the historical failure data for UF_6 cylinders, the long periods of time needed for sufficient water to accumulate in the cylinder, and the prohibition on storing cylinders enriched to >5wt% ^{235}U outdoors, the staff has reasonable assurance that occurrence of criticality in a cylinder with material enriched between 5 and 10wt% ^{235}U is extremely unlikely, and that an exemption to 10 CFR 70.24(a) for the cylinder storage yards is warranted.

5.3.7 Criticality Code Validation Report and Margin of Subcriticality

NRC staff's review of the criticality code validation report is contained in Section C.1.1 of this SER. NRC staff's review of the proposed margin of subcriticality (0.02 for abnormal conditions up to 5wt% ^{235}U; 0.05 for abnormal conditions from 5 to 10wt% ^{235}U) is contained in Section C.1.2 of this SER.

The requirements for validation and subcritical margin discussed in these sections together ensure the adequacy of the margin of subcriticality and ensure that an acceptable area of applicability will be maintained, as discussed in Sections 5.4.3.4.1(6)(1), (6)(2), and 5.4.3.4.1(10)(g) of NUREG-1520 (NRC, 2002). The description of the area of applicability and the requirement to seek NRC approval before exceeding the limits of the area of applicability described in the LA (USEC, 2006b) (see Section 5.3.4 of this SER) make a specific commitment to the acceptance criteria in Section 5.4.3.4.1(6)(3) of NUREG-1520 (NRC, 2002) unnecessary. In addition, the review conducted as described in Section 5.3.4.5.2 of this SER covers the acceptance criteria in Section 5.4.3.4.1(7)(a)-(j) of NUREG-1520 (NRC, 2002). The review conducted as described in Section C.1.1 of this SER covers the acceptance criteria in Section 5.4.3.4.1(8)(a)-(l) of NUREG-1520 (NRC, 2002). Also, the fact that the details of validation are described in the LA (USEC, 2006b) ensure that any significant changes would require an amendment, as discussed in Section 5.3.4.5.2 of this SER. This requirement makes a commitment to the acceptance criteria in Section 5.4.3.4.1(9) of NUREG-1520 (NRC, 2002) unnecessary. The determination of an upper safety limit as described in Section C.1.1 of this SER satisfies the acceptance criteria in Section 5.4.3.4.4(5) of NUREG-1520 (NRC, 2002).

The adequacy of the margin of subcriticality relative to the calculated bias is also discussed in the above section of Appendix C to this SER, satisfying the acceptance criteria in Section 5.4.3.4.1(10)(f) of NUREG-1520 (NRC, 2002). The subcriticality requirement discussed in C.1.2 of this SER also satisfies the acceptance criteria in Section 5.4.3.4.4(1) of NUREG-1520 (NRC, 2002). Because the minimum subcritical margins for normal and credible abnormal conditions are contained in the LA (USEC, 2006b), changing them would require NRC pre-approval. It is therefore not necessary for a specific commitment to pre approval, as stated in Section 5.4.3.4.4(4) of NUREG-1520 (NRC, 2002). This also satisfies the acceptance criteria in Section 5.4.3.4.4(8) of NUREG-1520 (NRC, 2002) regarding subcriticality of operations and margin of subcriticality for safety.

5.3.8 NCS in the ISA

NRC staff's review of the criticality safety aspects of the ISA Summary (USEC, 2006a) is contained in Section C.2 of this SER. This addresses the ISA acceptance criteria in Section 5.4.3.4.6 of NUREG-1520 (NRC, 2002) with regard to NCS (although performance of the ISA is discussed more generally in Chapter 3 of this SER).

Section C.2 of this SER discusses how the applicant implements the acceptance criteria in Section 5.4.3.4.6(1) of NUREG-1520 (NRC, 2002). The applicant's use of ANSI/ANS-8.1-1998 (ANSI/ANS, 1998) is discussed in Section 5.3.5 of this SER. There is no need for the applicant to use ANSI/ANS-8.10-1983 (ANSI/ANS, 1983b) to determine the consequences of a criticality accident, since the consequences are conservatively assumed to be high for the purposes of performing the ISA. Thus, meeting the acceptance criteria in Section 5.4.3.4.6(3) of NUREG-1520 (NRC, 2002) is unnecessary. In addition, there is no need for a specific requirement to use NCS methodologies and technical practices when evaluating NCS accident sequences in the ISA, as discussed in the acceptance criteria in Section 5.4.3.4.7(5) of NUREG-1520 (NRC, 2002). The technical practices applicable to all NCS evaluations are discussed in Section 5.3.4.5 of this SER.

5.3.9 NCS in the Emergency Plan

NRC staff's review of the criticality safety aspects of the Emergency Plan is contained in Section C.3 of this SER. This satisfies the acceptance criteria in Section 5.4.3.4.3(8)(b) of NUREG-1520 (NRC, 2002).

Part of emergency management includes determining whether a report to NRC is required in accordance with 10 CFR Part 70, Appendix A, as discussed in the acceptance criteria in Section 5.4.3.4.7(7)(a)–(d) of NUREG-1520 (NRC, 2002). These were not specifically reviewed as part of the NCS review, because the reporting requirements pertain to many types of events besides the occurrence of a criticality accident.

5.3.10 Baseline Design Criteria

10 CFR 70.64(a) states that an applicant must address certain BDC in the design of a new facility. Specifically, with regard to NCS, 10 CFR 70.64(a)(9) states:

> "Criticality control. The design must provide for criticality control including adherence to the double contingency principle."

The applicant's NCS Program, as described in Chapter 5 of the LA (USEC, 2006b), describes generally the measures that provide for criticality control. This satisfies the acceptance criteria with regard to BDC requirements in Section 5.4.3.1(8) of NUREG-1520 (NRC, 2002). Also, the applicant's ISA Summary (USEC, 2006a) describes in greater detail the specific features of the design that provide for criticality control for specific criticality accident sequences. The applicant's reference to ANSI/ANS-8.1-1998 (ANSI/ANS, 1998) includes following the DCP, the staff's review of which is contained in Section 5.3.5 of this SER. This requirement satisfies the acceptance criteria that no single credible event or failure can result in a criticality accident, in Section 5.4.3.4.2(2) of NUREG-1520 (NRC, 2002). In addition, this satisfies the double contingency acceptance criteria in Section 5.4.3.4.4(7) of NUREG-1520 (NRC, 2002) including preference for two-parameter control in 5.4.3.4.4(7)(a) (see Section 5.3.4.3 of this SER) and concurrence in 5.4.3.4.4(7)(b). The applicant has not indicated any exceptions to the DCP, and so the acceptance criteria in Section 5.4.3.4.4(7)(c) of NUREG-1520 (NRC, 2002) is not applicable. This also satisfies the acceptance criteria in Section 5.4.3.4.5(1) and 5.4.3.4.5(2) of NUREG-1520 (NRC, 2002) with regard to double contingency.

Based on this, the staff has determined that the applicant has met the BDC requirement in 10 CFR 70.64(a)(9) regarding criticality control.

5.4 EVALUATION FINDINGS

NRC staff reviewed the NCS information for the facility in accordance with Chapter 5 of NUREG-1520 (NRC, 2002). The staff concluded that the information submitted in Chapter 5 of the LA (USEC, 2006b) described an NCS program for the ACP that meets the applicable requirements of 10 CFR Part 70 (except 10 CFR 70.24, as discussed below), and is in accordance with the acceptance criteria of Chapter 5 of NUREG-1520 (NRC, 2002) or an acceptable alternative approach. In particular, the staff concluded that this program meets the baseline design criteria applicable to criticality safety, 10 CFR 70.64(a)(9).

The staff concluded that the information submitted in support of the requested exemption from 10 CFR 70.24 was sufficient to demonstrate that such an exemption for the UF_6 cylinder yards is authorized by law, will not endanger life or property or the common defense and security, and will otherwise be in the public interest, as required by 10 CFR 70.17(a). Based on the review documented in Chapter 5 of this SER, the staff concludes that construction and operation of the proposed ACP will provide for reasonable assurance of safety and the adequate protection of public health and safety from the consequences of a criticality accident.

5.5 REFERENCES

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6.0 CHEMICAL PROCESS SAFETY

The purpose of U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's chemical safety program and the design of the facility is to evaluate whether the application will adequately protect workers, public, and the environment during normal operations against chemical hazards of licensed material and its by-products. The chemical safety program and the facility's design must also protect against facility conditions and/or operator actions that can affect the safety of licensed materials and thus present an increased chemical risk.

6.1 REGULATORY REQUIREMENTS

The regulatory bases for the review are the general and additional contents of an application that addresses chemical process safety, as required by 10 CFR 70.22 and 70.65. In addition, the chemical process safety review should provide reasonable assurance of compliance with 10 CFR 70.61, 70.62, and 70.64.

6.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of chemical process safety for the proposed facility is contained in Chapter 6 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). This chapter is applicable in its entirety. The staff also uses NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities" (NRC, 1997a), and NUREG-1513, "Integrated Safety Analysis Guidance Document" (NRC, 2001), as guidance documents for this review. The acceptance criteria applicable to this review are contained in Section 6.4.3 of NUREG-1520 (NRC, 2002).

6.3 STAFF REVIEW AND ANALYSIS

NRC staff reviewed the license application (LA) (USEC, 2006c) and the Integrated Safety Analysis (ISA) Summary (USEC, 2006b) submitted by the applicant and considered the following areas:

1. Process Description;
2. Chemical Accident Sequences;
3. Chemical Accident Consequences;
4. Chemical Process items relied on for safety (IROFS);
5. Management Measures;
6. Emergency Management; and
7. Baseline Design Criteria (BDC).

The staff reviewed the applicant's responses to requests for additional information and ISA documents during on-site visits (NRC, 2004, 2005), as necessary, to have a better understanding of the process and safety requirements. The staff evaluated the information to determine if the facility's design complied with the BDC and defense-in-depth requirements specified in 10 CFR 70.64(a) and 70.64(b), respectively. Compliance with these regulations is discussed in more detail in Chapter 3, Appendix A, and Appendix B of this Safety Evaluation Report (SER). The staff's evaluation and general information about the American Centrifuge

Plant (ACP) process are summarized in the following sections.

6.3.1 Process Description

The applicant describes the gas centrifuge process in Chapter one of the LA (USEC, 2006c). The plant process is designed to enrich natural uranium hexafluoride (UF_6) in gaseous form by separating a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream enriched in the uranium-235 (^{235}U) isotope up to 10 weight percent ^{235}U and a tails stream depleted in the ^{235}U isotope. The process, entirely physical in nature, mechanically separates the isotopes using a fast rotating cylinder (centrifuge). This separation occurs because there is a difference in centrifugal forces between the isotopes of uranium since they have different molecular weights. No nuclear reactions nor significant chemical changes are expected to occur during normal operations. The feed, product, and tails streams are all in the form of UF_6. The nominal capacity of the facility will be about 3.5 million separative work units per year.

6.3.1.1 Gas Centrifuge Process

The ACP will be comprised of various buildings/facilities and areas on the U. S. Department of Energy (DOE) reservation in Piketon, Ohio. The ACP will primarily utilize existing buildings and facilities which were part of DOE's Gas Centrifuge Enrichment Plant, built in the early 1980s, but will also use newly constructed buildings and facilities. The facility the applicant proposes will be divided into the following operations:

- Receipt of UF_6;
- Feeding of UF_6 into the enrichment process;
- Enrichment processing using the cascade centrifuge machines;
- Enriched and depleted UF_6 withdrawal;
- UF_6 sampling to ensure it meets customer specifications;
- UF_6 product material transfer into customer cylinders;
- Loading of UF_6 cylinders for shipment to customers; and
- Handling of waste generated from the entire process.

Cylinders containing feed UF_6, cylinders containing enriched product, and customer shipping cylinders and overpacks, as well as new and/or cleaned empty cylinders, will be received on-site through the X-3346A building. The cylinders will be off-loaded, weighed, and transferred to the appropriate cylinder storage areas.

The major equipment used in the UF_6 feed process (X-3346 building) will be the feed ovens. Natural UF_6 will be delivered to the plant in American National Standards Institute (ANSI) N14.1, "Nuclear Materials - Uranium Hexafluoride - Packaging for Transport" (ANSI, 2001) standard type 14-ton international transit cylinders. Feed cylinders will be loaded into the electrically heated feed ovens; vented for removal of light gases, primarily consisting of air and hydrogen fluoride (HF); and heated to sublime the solid UF_6. Solid UF_6 left in the cylinder after the feed operation will be recovered by being "heeled" to a freezer-sublimer in the Burp System.

The cascade centrifuge machines will be contained in the Process Buildings (X-3001 and X-3002). UF_6 feed material will be supplied to the process from the Feed and Customer Service Building (X-3346) via heated interconnecting piping at subatmospheric pressure. Since

individual centrifuges will not be able to produce the desired product and tails concentration in a single step, the centrifuges will be grouped together in series and in parallel to form arrays known as cascades. A centrifuge consists of a vertical, cylindrically-shaped rotor that spins within an outer casing. As the UF_6 feed enters the cascade, it will be mixed with material already in the cascade and separated into enriched and depleted material streams.

Depleted UF_6 (tails) exiting the cascade will be transferred to the X-3356 building. Tails withdrawal will be accomplished through compression and direct desublimation into 10-ton or 14-ton cylinders. The major components that support the tails withdrawal operations will be the withdrawal (compression) trains, cold boxes, cold traps, assay spectrometers, and vents.

Product withdrawal will occur in the X-3356 building via desublimation into cold traps. Any "light gases" will be vented during this process. The cold traps will be heated to sublime the UF_6 which will be subsequently desublimed into 10-ton and 2.5-ton cylinders located in cold boxes. The filled source cylinders will then be moved to interim storage and subsequently moved to the X-3346 building sampling and transfer area.

UF_6 sampling and transfer operations will be carried out in the product operations area of the X-3346 building. The major components of these operations will be autoclaves, cold traps, and vents. The applicant will use the American Society for Testing and Materials (ASTM) C1052, "Standard Practice for Bulk Sampling of Liquid Uranium Hexafluoride," standard which requires that samples be taken from homogenized UF_6 (ASTM, 2001). The applicant will use ASTM C787-03, "Standard Specification for Uranium Hexafluoride Enrichment," and ASTM C996-04, "Standard Specification for Uranium Hexafluoride Enriched to Less than 5% ^{235}U," as part of its design which involves liquid UF_6 material during sampling, blending, and transfer operations (ASTM, 2003, and 2004). Electrically heated autoclaves will be used to liquify UF_6 in the source cylinders in order to facilitate the mixing of product and the transfer of liquid UF_6 to customer cylinders. To contain a UF_6 release, the autoclaves will be designed according to the American Society of Mechanical Engineers (ASME) standard, "Boiler for Pressure Vessel Code Section VIII, Pressure Vessels," 2004 (ASME, 2004a). The applicant will design process piping following ASME B31.1, "Process Piping," 2004, to minimize the potential for release of licensed material (ASME, 2004b).

Filled customer product cylinders, emptied feed cylinders, and other UF_6 cylinders will be prepared for shipment in the X-3346A building. These cylinders will meet the ANSI N 14.1 standard (ANSI, 2001). Section 1.1.3.3.2 of the LA (USEC, 2006c) contains additional information on the operations performed in this building.

Depleted UF_6 is handled in the ACP. No process waste water will be expected to be discharged from the liquid effluent tanks. Each process area vent system in the Process Buildings (X-3001 and X-3002), Feed and Customer Service Building (X-3346), Sampling and Transfer Area (X-3346), Product and Tails Withdrawal Building (X-3356), and the Recycle/Assembly Facility (X-7725) will have gas flow monitoring and analytical instrumentation to continuously sample, monitor, and alarm if UF_6 is detected in the effluent gas stream.

The entire enrichment process system will operate at subatmospheric pressure with the exception of the sampling and blending process. This safety feature will help to minimize gaseous releases of UF_6 and HF since the leakage of material will typically be inward to the system. During sampling and blending operations, UF_6 will be liquified within an autoclave that

will provide the heating required to homogenize the material for sampling and/or blending. The cylinders containing liquid UF_6 will be designed following the ANSI N 14.1 (ANSI, 2001) standard while the autoclaves will be designed following the ASME Boiler for Pressure Vessel Code Section VIII (ASME, 2004a) standard. The cylinder and the autoclave will serve as the primary and the secondary containment, respectively, of UF_6 in liquid state.

The applicant provides more detailed discussion of the processes described above in the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b).

6.3.1.2 Chemical Process Inventories

The applicant stated in section 6.1 of the LA (USEC, 2006c) that the inventories of chemicals will be maintained below the threshold quantities set forth by the Occupational Safety and Health Administration (OSHA) Process Safety Management (PSM) standard, 29 CFR 1910.119, and the U. S. Environmental Protection Agency (EPA) Risk Management Program standard, 40 CFR Part 68. UF_6 and its reaction compounds will comprise the greatest quantities of hazardous materials at the ACP.

The applicant expects to possess source material and special nuclear material as part of the process and byproduct radioactive materials as calibration sources. The majority of the calibration sources will be sealed and will not constitute a chemical hazard in the facility.

Chemical Safety Control Strategy

The chemical safety control strategy will involve: (1) the identification of chemicals used in the facility; (2) maintaining the chemical listing up to date; and (3) the review of potential hazards. The chemical hazards are addressed in the ISA Summary (USEC, 2006b) and by the applicable Industrial Hygiene Safety (IHS) programs. The chemical inventory will be identified through a process that will involve:

- Purchase requisition reviews;
- A listing of chemicals used;
- Material Safety Data Sheet (MSDS) library, updates, and distribution services to the plant; and
- Identification of new chemicals for the review process.

Formal requests for engineering services will be required for modifications to existing systems. The request process will provide a mechanism that will identify new or revised chemical usage and processes and/or associated possible logistics that require Engineering involvement. Contractor Safety and Health Plans will be reviewed to identify the presence or use of toxic or hazardous materials the contractor plans to bring on-site. Chemicals other entities use on the DOE reservation will be covered under a shared site agreement.

6.3.1.3 Hazardous Chemicals and Chemical Interactions

The main chemical hazard present in the ACP will be UF_6 and its by-products. Any UF_6 that is released to the environment will react exothermically with the water vapor present in air, producing solid uranyl fluoride (UO_2F_2) and HF gas. The chemical form of the products of this reaction will depend on the temperature and the relative humidity of the air at the time of the

release (see Equation 1).

$$UF_{6(gas)} + 2H_2O_{(vapor)} \rightarrow UO_2F_{2(solid)} + 4HF_{(gas)} + Heat \quad \text{Eq. 1}$$

At room temperature, hydrated forms of UO_2F_2 and HF can be produced depending on the relative humidity in the air (see Equation 2) (USEC, 1999).

$$UF_{6(gas)} + (2+4x)H_2O_{(vapor)} \rightarrow UO_2F_2 \cdot 2H_2O_{(solid)} + 4HF \cdot xH_2O_{(gas+vapor)} + Heat \quad \text{Eq.2}$$

Usually, UO_2F_2 compounds are deposited or precipitated close to the point of the release.

Some chemicals expected to be used at the ACP include lubricant oil, gases (e.g., nitrogen), diesel fuel, and chemical trap media. The applicant will implement industrially-recognized standards in completing the design of the ACP and has documented an ISA that will bound the design of the ACP.

Lubricant Oils

Besides the interaction with moisture, UF_6 can react exothermically with hydrocarbons. Gaseous UF_6 can react with hydrocarbons to form a black residue of uranium-carbon compounds. The reaction rate of this chemical reaction is accelerated when UF_6 is in liquid phase and may lead to an explosive environment (USEC, 1999).

In Section 1.1.10 of the LA (USEC, 2006c), the applicant states that, as much as possible, the ACP will be designed and constructed to use pumps and compressors that do not require the use of oil. However, where lubricants are needed for equipment that could contact UF_6 gas, the applicant will use lubricants compatible with UF_6 and HF. The preferred lubricants are polyfluoropolyethers (PFPE), which are commonly referred to by manufacturers' trade names of Fomblin and Krytox. Fomblin and Krytox are inert, fully fluorinated, and do not react with UF_6 under any operating conditions. These lubricants have minimal flammability and toxicity concerns.

Hydrocarbon based lubricants may be used if process equipment cannot achieve the desired performance parameters using PFPE lubricants. The amounts of hydrocarbon based lubricants in these instances will be small enough to minimize the criticality and combustible loading concerns.

Small quantities of uranium compounds and traces of hydrocarbons may be contained in the Fomblin and Krytox. The UF_6 degrades in the lubricants or reacts with trace hydrocarbons to form crystalline compounds - primarily UO_2F_2 and uranium tetrafluoride (UF_4) particles that gradually thicken the lubricants and reduce equipment capacity. As a result, the lubricants will require periodic replacement.

Chemical Trap Media

The applicant will be using chemical traps containing alumina and soda lime pellets to capture small amounts of UF_6 and HF. The evacuation system and the purge system will be the major vacuum systems used in this facility. Both subsystems will vent to the atmosphere after removing trace amounts of UF_6 and HF from operational activities in the chemical traps. The sampling systems will also vent through chemical traps. The efficiency of the traps will depend on the concentration of UF_6 or HF and the flow rates incoming from operations.

Materials of Construction

The applicant states in Section 1.1.9 of the LA (USEC, 2006c) that the facility will be designed to ensure that piping and other equipment can maintain a minimum wall thickness during the operations life of the plant (30 years). Corrosion rates are not expected to exceed 0.0025 millimeters per year depending upon materials of construction, equipment configurations, and flow rates.

Section 1.1.9 of the LA (USEC, 2006c) and Section 3.14 of the ISA Summary (USEC, 2006b) provide examples of the types of construction materials and the conditions under which they would optimally perform under the expected conditions. For different areas of the facility, the applicant provided examples of construction materials that would be suitable for the expected temperature, pressure, and corrosive conditions. An example is the use of steel in UF_6 cylinders. Although steel will corrode in the presence of UF_6, the design of the cylinders compensates by increasing the wall thickness. In addition, operational requirements require periodic retesting of the cylinders to ensure that the residual wall thickness is still adequate even under high temperature conditions experienced during cylinder heating. In Section 1.4.2 of the LA (USEC, 2006c), the applicant committed to ANSI N14.1 (ANSI, 2001) which contains these requirements. Materials proposed for use in other parts of the facility are chosen to ensure that the effects of corrosion and erosion are minimized. The applicant also addressed the selection of seals and gaskets and the selection of soldering and brazing alloys.

During the Operational Readiness Review conducted in accordance with 10 CFR 70.32(k), the staff will evaluate the applicant's final selection of lubricant oils, chemical trap media, and materials of construction.

6.3.1.4 Process Description Conclusion

The staff finds that the applicant has provided process descriptions that are sufficiently detailed to allow an understanding of the chemical process hazards. The information that the applicant provided, as described above, meets the guidance in Section 6.4.3.1(1) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The staff finds that the applicant has provided descriptions of the approach employed for adequately maintaining safety in normal operations. The information that the applicant provided, as described above and in Section 6.3.4.2.2 of this SER, meets the guidance in Section 6.4.3.2(1) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The staff also finds that the applicant has provided a description of the chemical hazards that could result from potential chemical interactions. The information provided by the applicant meets the guidance for the evaluation of potential process chemical interactions in Section 6.4.3.1(1) of NUREG 1520 (NRC, 2002), and is therefore acceptable.

6.3.2 Chemical Accident Sequences

The ISA Summary (USEC, 2006b) and Chapter 3 of the LA (USEC, 2006c) discuss the screening criteria the applicant used to identify chemicals for further analysis in the hazard evaluation. The methodology used to perform the hazards evaluation phase is also discussed in these chapters.

As part of the ISA process, the applicant performed a screening of the chemicals by using 40 CFR Part 68, 29 CFR 1910.119, or the threshold planning quantity listed in 40 CFR 355 as guidance in order to identify chemicals that can be present in threshold quantities. The chemical would be screened out from the Hazards Evaluation phase if it met the following criteria (USEC, 2006c):

- It was not present in threshold quantities;
- It was analyzed and did not represent a significant hazard;
- It was a sample, laboratory scale quantity, and/or designated as a laboratory quantity (under ten pounds); or
- The industry and/or the general public commonly use the material, it did not present an airborne concern, and/or the material does not cause an acute exposure by inhalation.

The applicant also considered the potential for a chemical to be an initiator to an event. In developing and documenting the accident sequences, the applicant also addressed the likelihood of the accidents to occur. The accident sequence analysis and the staff's evaluation of the likelihoods proposed by the applicant are further discussed in Appendix A of this SER.

The ISA Summary (USEC, 2006b) includes potential chemical accident sequences and identifies selected controls that either prevent or mitigate the consequences to an acceptable level. The chemical accident sequences covered potential accidents in the: (a) Process Buildings (X-3001 and X-3002); (b) Process Support Building (X-3012); (c) Cylinder Yards (X-7746N, X-7746S, X-7746E, X-7746W, X-7756S, X-745G-2, and X-745H); (d) Interconnection Process Piping (X-2232C); (e) Feed and Customer Service Building (X-3346); (f) Sampling and Transfer Area (X-3346); (g) Product and Tails Withdrawal Building (X-3356); (h) Recycle/Assembly Facility (X-7725); (i) Centrifuge Training and Test Facility (X-7726); (j) Interplant Transfer Corridor (X-7727H); and (k) Feed and Product Shipping and Receiving Building (X-3346A). The accident sequences covered the range of events that could result in loss of confinement of UF_6, the hazardous chemicals produced from UF_6 (i.e., HF, UO_2F_2, and interaction with contaminants in cylinders), and the potential interactions of UF_6 with the process materials of construction (see Section 6.3.1.3 of this SER). The accident sequences addressed both intermediate and high consequence events. The staff performed a risk-informed review of selected chemical accident sequences and also performed two vertical slice reviews. Chemical accident sequences are discussed in more detail in Chapter 3, Appendix A, and Appendix B of this SER. The staff concludes that the applicant has identified appropriate chemical accident sequences, including appropriate accident likelihoods, based on the applicant's use of a combination of approved hazards analysis methods (PHA and WI/CL methods) to identify those sequences and the staff's review of selected chemical accident sequences. The information provided by the applicant, as described above and in Chapter 3, Appendix A, and Appendix B of this SER, meets the accident sequence and likelihood guidance in Section 6.4.3.1(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

6.3.3 Chemical Accident Consequences

The ISA Summary (USEC, 2006b) addresses the chemical quantitative risk levels used in determining the impact of potential accidents on the workers and the public. The applicant has chosen to use the Emergency Response Planning Guidelines (AIHA, 2004) as consequence action levels in order to evaluate accidents associated with chemical events involving HF, in a manner consistent with NUREG-1520, Table A-5, "Consequence Severity Categories Based on 10 CFR 70.61" (NRC, 2002). For events involving soluble uranium compounds, the applicant has defined action levels to be in agreement with the performance requirements using the intake of soluble uranium in milligrams (USEC, 2006c). These consequence levels are also described in Chapter 3, Appendix A, and Appendix B of this SER. The staff finds this approach to be consistent with the guidance in Section 6.4.3.1(6) of NUREG-1520 (NRC, 2002), and is therefore acceptable for the determination of compliance with the performance criteria of 10 CFR 70.61.

The applicant determined the accident consequences using: (1) source term calculations; (2) existing safety documentation; and/or (3) qualitative assessment. The applicant's Hazard Analysis team used its discretion, expertise, and knowledge of the process during the hazards assessment process. The applicant also used available documentation from DOE, NRC's guidance, and information available in literature as part of the accident consequences analysis. The applicant used the HG System, which is an acceptable atmospheric dispersion model evaluated in NUREG/CR-6481, "Review of Models Used for Determining Consequences of UF_6 Releases" (NRC, 1997b), to calculate the concentration of UF_6 and its reaction products in some fire events. The applicant also used an atmospheric dispersion factor (c/Q) correlation and an estimated source term to calculate the concentration of UF_6 and its reaction products (USEC, 2006b). The source term was estimated using a five factor formula consistent with the guidance in NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook" (NRC, 1998):

ST = MAR x DR x ARF x RF x LPF

- ST = Source term
- MAR = Material at risk, amount of hazardous material available to be acted upon by a given physical stress;
- DR = Damage ratio, fraction of material actually impacted by the accident;
- ARF = Airborne release fraction, the coefficient to estimate the amount of material suspended in air as an aerosol, vapor, or gas and thus available for airborne transport because of physical stress from a given accident;
- RF = Respirable fraction, fraction of airborne radionuclides or chemical aerosols that can be transported through air and inhaled into the human respiratory system;
- LPF = Leak path factor, fraction of radionuclides or chemical aerosols in the air transported through some confinement, deposition, or filtration mechanism.

The applicant used the ARF and RF vales from DOE-HDBK-3010-94, "Airborne Release Fractions and Respirable Fractions for Use with DOE Non-Reactor Nuclear Facilities" (DOE, 1994). These values are the source of the values found in NUREG/CR-6410 for the ARF and RF values. The applicant used different approaches to estimate the MAR since it would depend on the mechanism of the release and the physical state of UF_6 in the process. Source term values are included in Appendix D of the ISA Summary (USEC, 2006b) and ISA documentation.

The staff finds that the applicant has identified and used appropriate techniques and valid assumptions in estimating the consequences from analyzed chemical accident sequences and that the consequences have been conservatively estimated. The information provided by the applicant provided, as described above, meets the guidance in Sections 6.4.3.1(3), 6.4.3.1(4), and 6.4.3.1(5) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The staff performed two vertical slice reviews of a representative sample of the potential accident sequences in the ACP. NRC staff also reviewed selected accident sequences and the process descriptions and process flow diagrams provided in Appendix E of the LA (USEC, 2006c) as part of the vertical slice reviews. During these reviews, in addition to reviewing ISA supporting documentation, the staff toured the Lead Cascade facility and areas of the Portsmouth Gaseous Diffusion Plant that have similar processes in operation. Details on the staff's evaluation of selected accident sequences are presented in Appendices A and B of this SER.

Based on the review of the ISA Summary (USEC, 2006b), on-site visits to evaluate supporting ISA documentation, and the review of selected accident sequences, the staff concludes that the applicant has adequately identified the consequences of the accident sequences involving the chemical hazards of licensed materials and hazardous chemicals produced from licensed material. The information the applicant provided, as described above, meets the accident consequence guidance in Section 6.4.3.1(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

6.3.4 IROFS and Management Measures

6.3.4.1 Chemical Process IROFS

The accident consequences and the need for IROFS were determined by applying the criteria described in 10 CFR 70.61. IROFS (engineering and administrative controls) were selected to prevent or mitigate the consequences to the public, workers, and the environment.

Chapter 7 of the ISA Summary (USEC, 2006b) describes the safety functions of all identified IROFS and the specific accident sequences that take credit for each IROFS. The staff reviewed selected chemical accident sequences and their respective process IROFS for the ACP (see Appendices A and B of this SER). NRC staff reviewed the listed IROFS and the process descriptions and process flow diagrams provided in Appendix E of the LA (USEC, 2006c) and how the IROFS would function to prevent or mitigate the consequences of the identified accident sequences. Based on the review of ISA documentation and on-site visits to the Lead Cascade and similar processes to be used in the ACP, the staff concludes that the applicant has adequately identified chemical process IROFS to prevent and/or mitigate the consequences of accident sequences involving the chemical hazards of licensed materials and hazardous chemicals produced from licensed material. The information the applicant provided, as described above and in Appendices A and B of this SER, meets the guidance in Section 6.4.3.2(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

6.3.4.2 Management Measures

After selecting the IROFS, management measures were selected to ensure that IROFS would be available and reliable to perform their safety function when required. A Quality Level (QL)

will be assigned to each IROFS based on the following criteria described in the Quality Assurance Program Description (QAPD) (USEC, 2006d):

QL-1 A single IROFS that prevents or mitigates a high consequence event.

QL-2 Two or more IROFS that prevent or mitigate a high consequence event; or one or more IROFS that prevents or mitigates an intermediate consequence event.

QL-3 Any item other than QL-1 and QL-2; QL-3 items are controlled in accordance with standard commercial practices.

The programmatic requirements applied to QL-1 and QL-2 IROFS are discussed in the QAPD (USEC, 2006d). In the QAPD, the applicant commits to meeting the requirements of 10 CFR 70.64(a)(1) and selected portions of the guidelines of ANSI/ASME Quality Assurance standard NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities" (ANSI/ASME, 1994).

The management measures the applicant proposes for the ACP are similar to the Lead Cascade and the Portsmouth Gaseous Diffusion Plant. The Lead Cascade and ACP are similar facilities in terms of gas centrifuge operation. The applicant states in Chapter 11 of the LA (USEC, 2006c), that management measures will be applied to IROFS, to provide reasonable assurance that the IROFS will be available and reliable to perform their functions when needed.

The following sections discuss the management measures the applicant proposes regarding the chemical safety program. Chapter 11 of this SER provides an evaluation of the management measures applied to the ACP.

6.3.4.2.1 Configuration Management (CM)

Engineering will administer the CM program. The CM program will address the organizational structure, administrative controls, procedures, and policies to ensure that accurate and current facility design documentation is maintained that matches the physical configuration of the facility.

The applicant will perform initial assessments of the program to identify, address, and document deficiencies and/or recommendations before the operational phase of the facility. The CM program is described in Section 11.1 of the LA (USEC, 2006c) and Chapter 11 of this SER.

6.3.4.2.2 Procedures

The applicant will prepare procedures in accordance with a formal procedure system as described in Section 11.4 of the LA (USEC, 2006c). Activities involving nuclear material and/or IROFS will be conducted in accordance with the appropriate approved procedures.

The applicant subleases certain buildings and facilities on the DOE reservation, which hosts other entities with their own chemical safety programs. Section 6.2.2.1.2 of the LA (USEC, 2006c) states that the applicant and DOE will share information concerning hazardous chemicals used by each entity under the shared-site agreement.

The safety and health program procedures will implement the IHS programs to be used for

chemical safety within the facility. The IHS program will generally include procedures such as:

- Lockout/Tagout;
- Hazard Communication;
- Confined Space Entry;
- Safety and Health Work Permit;
- Hot Work Permit;
- Personal Protective Equipment;
- Signs/Labeling/Tagging; and
- Safety Training.

The safety and health program procedures will follow the Records Management and Document Control Program requirements to ensure that QAPD records and documents will be properly managed.

6.3.4.2.3 Training

The Production Support Manager will be responsible for ACP employee training. The applicant will have prerequisite and periodic training requirements for initial and continued job qualification for personnel involved in operations, emergency response, maintenance, and/or management activities. Personnel who perform duties for a particular or related activity involving a chemical safety system will receive specific safety training related to the task.

Facility personnel will be trained to recognize and overcome safety hazards, such as chemical hazards, that may be encountered in the work place under normal or abnormal conditions. The facility will also adopt the "See and Flee" policy for the ACP, similar to the policy currently used at the Lead Cascade, which is based on the Portsmouth Gaseous Diffusion Plant's "See and Flee" policy. This policy specifies that personnel immediately move away from the release area to a safe location. In Section 6.2.2.5 of the LA (USEC, 2006c), the applicant states that mitigating actions may be performed before or during evacuation if these actions do not hinder safe egress.

The Production Support Manager will have the overall responsibility for employee training while the Site Technical Representative will be the liaison between the applicant and the contractor. The contractor or the contractor-designated Safety and Health Officer will be in charge of the internal training for the contractors. Contractors will receive facility-specific training before starting any work. The applicant will approve the contractor's Safety and Health Plan and will oversee construction activities that can interface with chemical systems.

6.3.4.2.4 Maintenance and Inspection

Engineering will review and approve maintenance and inspection requirements for chemical systems and components by taking into consideration manufacturer recommendations, and the functions of IROFS described in the ISA Summary (USEC, 2006b).

The maintenance program will cover:

- Calibration and Inspection - The applicant will consider manufacturer recommendations, operating nature and environment, the ISA Summary (USEC, 2006b), and prior operating experience in developing calibration and inspection requirements.
- Maintenance Work Packages - The applicant will provide the necessary technical and safety guidance for maintenance work activities in maintenance work packages that will be applicable to chemical systems and equipment.
- Preventive Maintenance and Quality Consideration - Sections 6.2.2.3.3 and 11.2.5 of the LA (USEC, 2006c) state that preventive maintenance will use manufacturer recommendations as guides to prevent failures, facilitate performance, and maintain or extend the life of the equipment. Inspection and testing will be based on a graded quality approach. The independent overview programs will include:

    - Procurement Quality Requirements;
    - Construction Inspection;
    - Testing and Pre-operational Inspection;
    - Pressure Vessel Inspection;
    - Crane Inspection;
    - Pre-operational Safety Review and Pre-startup Safety Review Programs; and
    - Plant Safety Review Committee.

The applicant states in Section 6.2.2.3.3 of the LA (USEC, 2006c) that the deficiencies associated with maintenance activities will be dispositioned in accordance with the QAPD and Corrective Action program.

6.3.4.2.5 Chemical Process Safety Records

The applicant states in Section 11.6.2 of the LA (USEC, 2006c) that it will maintain records of auditable documentation related to abnormal events, investigations, and root cause analyses. The applicant states in Section 6.3 of the LA (USEC, 2006c) that the records of chemical releases and documentation related to chemical process safety will be maintained as described in Section 11.7.1.5 of the LA (USEC, 2006c). Section 11.7.1.5 of the LA (USEC, 2006c) states that the manager of the organization originating the records or its designee will develop a retention time schedule with specific time periods for retention of these records. Record keeping and document control are discussed in more detail in Chapter 11 of this SER.

6.3.4.2.6 Audits and Assessments

Qualified ACP personnel will perform periodic audits and assessments, to verify safety during operations. These personnel will also verify that plant operations are conducted in accordance with the regulatory requirements and LA (USEC, 2006c) commitments (e.g., chemical safety). The applicant states in section 11.5.1 of the LA (USEC, 2006c) that the organizations in charge of performing the audits will be independent from ACP operations activities. The applicant will document and report the results from audits and assessments to the pertinent authorities. The applicant will make provisions for reporting and corrective actions, where warranted, in accordance with the ACP Corrective Action program.

6.3.4.2.7 Incident Investigation

Identification, reporting, and incident investigations are described in Section 11.6 of the LA

(USEC, 2006c). The incident investigation and reporting program will be conducted in accordance with the ACP procedures and the requirements in 10 CFR 70.50 and 70.74.

The level of investigation will depend on the severity of the event in terms of safety and the regulatory requirements involved. The deficiencies in the performance of the program will be addressed in the ACP Corrective Action program.

Occupational illness and/or injury investigations related to chemical safety events will be conducted in accordance with OSHA requirements.

6.3.4.3 IROFS and Management Measures Conclusion

The staff finds that the applicant has adequately identified the administrative and engineered controls (IROFS) to prevent or mitigate chemical process risks at the proposed facility. The information the applicant provided, as described above and in Appendices A and B of this SER, meets the guidance in Section 6.4.3.2(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The staff finds that the applicant has provided a sufficient description of how IROFS and management measures will be graded and how such grading is commensurate with the reduction in risk that the IROFS are designed to achieve. The information the applicant provided, as described above and in Chapter 11 of this SER, meets the guidance in Sections 3.4.3.2(6) and 6.4.3.2(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The staff finds that the applicant has provided a sufficient description of its procedures to ensure the reliable operation of engineered controls and that administrative controls will be correctly implemented. The information the applicant provided, as described above and in Chapter 11 of this SER, meets the guidance in Section 6.4.3.2(3) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

6.3.5 Emergency Management

The applicant's emergency management program is discussed in more detail in Chapter 8 of this SER and in the applicant's Emergency Plan (USEC, 2006a) submitted with the ACP LA (USEC, 2006c). Personnel with emergency response assignments will be trained to respond to chemical and operational upsets per 29 CFR 1910.120(q). Section 7.2.3 of the Emergency Plan (USEC, 2006a) states that the applicant will offer training to off-site emergency response personnel who may be called to respond to emergencies at the DOE reservation.

In case of a chemical release and/or a minor chemical spill, operators will be expected to evacuate the immediate area by following the "See and Flee" policy. The policy specifies that the employees must move to a safe location. Emergency response personnel are responsible for mitigating the release. Specific areas in the ACP will also be equipped with alarm systems in order to alert operators of UF_6 releases. The staff reviewed the ISA Summary (USEC, 2006b) and determined that these actions are consistent with the accident sequences and mitigating IROFS identified in the ISA Summary (USEC, 2006b) and are consistent with the actions described in Chapter 8 of NUREG 1520 (NRC, 2002).

In Chapter 10 of the Emergency Plan (USEC, 2006a) for the ACP, the applicant states that the

facility will comply with the EPA Emergency Planning and Community Right-to-Know Act of 1986, Title III, as required by 10 CFR 70.22(i)(3)(xiii). Some aspects related to this topic are also discussed in Section 6.2.2.11 of the LA (USEC, 2006c). The applicant states that the MSDSs will be maintained in a central location within the ACP and will be available at all times to plant personnel. The MSDSs will be routinely updated by Industrial Hygiene personnel. Contractors will also maintain hard copies of MSDSs at the job site. The applicant also agreed to maintaining MSDSs on-site and to share this information with on-site and off-site organizations that may be expected to respond to an emergency at the ACP (USEC, 2006c).

The staff finds that the applicant has provided reasonable assurance that measures to mitigate the consequences of accident sequences identified in the ISA Summary (USEC, 2006b) are consistent with actions described in Chapter 8 of NUREG 1520 (NRC, 2002). The information the applicant provided, as described above, meets the guidance in Section 6.4.3.1(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

6.3.6 BDC

The applicant provides design bases information for chemical process safety IROFS identified for the proposed facility in the ACP LA (USEC, 2006c) and ISA Summary (USEC, 2006b). For chemical protection, 10 CFR 70.64(a)(5) states:

> "Chemical protection. The design must provide for adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material."

The main chemicals of concern at the ACP will be UF_6 and the reaction products when it reacts with water (see Section 6.3.1.3 of this SER). Details of the design and safety features can be found in the LA (USEC, 2006c) and the ISA Summary (USEC, 2006b). The applicant will have IROFS in place to prevent different types of accidents and to mitigate consequences to the public and the workers. The applicant has also identified additional safety controls that will constitute defense-in-depth in the facility's design (see Appendix A and Appendix B of this SER). In describing the application of the chemical protection BDC to the proposed facility design, the applicant did not propose any facility-specific or process-specific relaxations or additions to the BDC.

The staff reviewed the applicant's proposed design of the gas centrifuge uranium enrichment facility contained in the ISA Summary (USEC, 2006b) and in Chapters 1 and 6 of the LA (USEC, 2006c). The staff noted that the uranium enrichment process is basically a physical process that separates the U^{235} isotope from the U^{238} isotope based on their mass difference. The entire process, with exception of the sampling and transfer of product into customer cylinders, will be conducted under a vacuum. As a result, any process leak would result in air-inleakage into the system, increasing system pressure to atmospheric pressure. Under normal process temperatures, as the system pressure approaches one atmospheric pressure, the gaseous UF_6 will desublime directly into the solid phase. Furthermore, the process design involves limited inventories of UF_6. Any UF_6 that could escape through a gaseous system breach will be limited by the available inventory and molecular diffusion since the process will be conducted with gaseous UF_6 in a vacuum (USEC, 2006b). The chemical behavior of UF_6 and its reaction products are such that most of the uranium bearing material is likely to accumulate near the process breach. As mentioned in the previous paragraph, the applicant proposes a safety

strategy in which it will have IROFS in place and will implement defense-in-depth practices as part of the design of the facility. Based on the operational mode of the centrifuges and the enrichment process, the limited inventories of licensed material contained in portions of the enrichment process, and the applicant's proposed safety strategy, the staff concludes the proposed design basis will provide adequate protection against chemical risks.

The applicant will use internationally recognized codes and standards as part of the design of the ACP. The staff notes that the chemical performance requirements of 10 CFR 70.61 can only be exceeded at the ACP through the loss of confinement of the licensed material. The licensed material (feed, product, and tails) will be stored and transported in ANSI/N14.1 (ANSI, 2001) qualified cylinders. These cylinders will serve as the primary containment for UF_6. Licensed material in the gaseous state in the process will be contained by process vessels and piping under vacuum. Licensed material in the liquid state will be contained in autoclaves and piping. The autoclaves will be designed to the ASME standard, "Boiler and Pressure Vessels Code Section VIII," 2004 (ASME, 2004a) and will act as a secondary containment. All process piping will meet the appropriate ASME code (i.e., ASME B31.1, "Process Piping," 2004). The applicant has stated that the materials of construction will be compatible with UF_6 and HF to ensure structural integrity of the process equipment. The applicant will use ASTM standards as part of the design bases for the liquid transfer operations (see Section 6.3.1.1 of this SER).

The staff reviewed the results of the applicant's PHA-WI/CL analysis as discussed in Chapter 3 of this SER. The applicant combined these methods which are used in the chemical industry to identify safety issues and are identified as an acceptable method in NUREG-1513, "Integrated Safety Analysis Guidance Document, " (NRC, 2001). As applied to the gas centrifuge uranium enrichment process, the PHA-WI/CL considered facility and external hazards that could breach the process and release licensed material and hazardous chemicals produced from licensed materials. The results of the applicant's ISA are presented in Appendix C of the ISA Summary (USEC, 2006b). The table contains information concerning the accident sequences identified as a result of the PHA-WI/CL. Appendix F presents the unmitigated risk of sequences that could exceed the performance requirements of 10 CFR Part 70, Subpart H, and the IROFS applied to prevent and/or mitigate the accident sequence. The staff reviewed selected consequence accident scenarios to confirm that chemical events that could exceed the performance requirements of 10 CFR Part 70 were addressed.

Based on the above, the staff concludes that the information the applicant provided and the applicant's proposed design meets the guidance in Section 6.4.3.3 of NUREG-1520 (NRC, 2002), provides for adequate protection against chemical risks produced from license material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed materials, and meets the requirements of 10 CFR 70.64(a)(5).

6.4 EVALUATION FINDINGS

The staff evaluated the application using the criteria previously listed. Based on the review of the LA (USEC, 2006c), NRC staff has concluded that the applicant has described and assessed accident consequences that can result from the handling, storage, or processing of licensed materials that can potentially have significant chemical consequences and effects. The applicant has prepared a hazard analysis that identifies and evaluates those chemical process hazards and potential accidents and established safety controls providing reasonable assurance of safe facility operation. To ensure that the performance requirements in 10 CFR

Part 70 are met, the applicant has stated that controls are maintained, available, and reliable to perform their safety-related functions when needed. The staff has reviewed a representative sample of the safety controls and the applicant's plan for managing chemical process safety and finds them acceptable.

The staff concludes that the applicant's plan for managing chemical process safety and chemical process safety controls meet the requirements of 10 CFR Part 70 and provides reasonable assurance that the public health and safety, and the environment, will be protected.

6.5 REFERENCES

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(USEC, 2006c) USEC Inc. (USEC). “License Application for the American Centrifuge Plant in Piketon, Ohio,” Revision 18, September 2006.

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7.0 FIRE SAFETY

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's fire safety program is to evaluate whether the application 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 radiological and chemical consequences of the fires and will institute suitable safety controls to protect workers, the public, and the environment.

7.1 REGULATORY REQUIREMENTS

The regulatory basis for the fire safety review should be the general and additional contents of application, as required by 10 CFR 30.33, 40.32, 70.22, and 70.65. In addition, the fire safety review should focus on providing reasonable assurance of compliance with 10 CFR 70.61, 70.62, and 70.64.

7.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the fire safety description section of the license application (LA) (USEC, 2006b) is contained in Chapter 7 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). This chapter is applicable in its entirety. The acceptance criteria applicable to this review are contained in Sections 7.4.3.1 through 7.4.3.5 of NUREG-1520 (NRC, 2002).

7.3 STAFF REVIEW AND ANALYSIS

This chapter addresses the staff's review of facility fire protection, including fire safety management measures, fire hazards analyses (FHAs), facility fire protection, process fire safety, and fire safety and emergency response, as presented in the LA (USEC, 2006b).

The staff review of process fire hazards and special hazards, fire related accident scenarios, and fire related items relied on for safety (IROFS) is contained in Appendix D, "Fire Safety," of this Safety Evaluation Report (SER). The appendix evaluates information contained in the Integrated Safety Analysis (ISA) Summary (USEC, 2006a) which is marked as "Export Controlled Information" and is designated by the NRC as "Official Use Only-DOE/NOFORN."

The applicant is proposing to construct its facility at the Portsmouth Gaseous Diffusion Plant (PORTS) in Piketon, Ohio, using some existing structures and constructing some others. The existing buildings, built for the Gas Centrifuge Enrichment Plant, were designed and constructed in the 1970s and 1980s to meet the applicable codes and standards applicable at the time. The LA (USEC, 2006b) states that new buildings/facilities will meet codes and standards applicable at the time of design. This will assure that the new buildings will meet or exceed the same standards for fire safety as the existing buildings.

The Fire Safety program and the American Centrifuge Plant (ACP) were reviewed to determine applicability and level of compliance with National Fire Protection Association (NFPA) 801 (NFPA, 2003e) and applicable daughter standards. Some ACP buildings/facilities do not meet

NFPA 801 (NFPA, 2003e) and applicable daughter standards because they were built or established under earlier versions or different codes and standards applicable at the time of construction and installation. The standards applicable to these ACP buildings/facilities will be documented during the baseline configuration assessment effort as described in Section 11.1 of the LA (USEC, 2006b). Table 7-1 of the LA (USEC, 2006b) lists the fire codes and standards considered by the applicant to be applicable to the facility.

Table 7-1
Applicable National Fire Protection Association Codes and Standards[1]

Code No. and Revision Year	Title	Exceptions
IEEE 484, 2002	IEEE Recommended Practice for Installation Design and Implementation of Vented Lead-Acid Batteries for Stationary Applications	The applicant will satisfy the provisions of this standard.
NFPA 10, 2002	Standard for Portable Fire Extinguishers	The provisions of this standard were used as guidance in determining the size, selection, and distribution of portable fire extinguishers. The applicant will use this standard for modifications to the facility except as justified through documentation.
NFPA 13, 2002	Standard for Installation of Sprinkler Systems	The provisions of this standard were used as guidance for the design and installation of wet and dry pipe automatic sprinkler systems. In addition, ACP facility meets the definition of Ordinary Hazard Occupancies (Group 1) as stated in this standard and the fire protection systems meets or exceeds the sprinkler discharge requirements for this type of occupancy. The applicant will use this standard for modifications to the facility except as justified through documentation.

[1] From Table 7.1-1 of the LA (USEC, 2006b)

Code No. and Revision Year	Title	Exceptions
NFPA 15, 2001	Standard for Water Spray Fixed Systems for Fire Protection	The applicant will use this standard for modifications to the facility except as justified through documentation.
NFPA 25, 2004	Standard for the Inspection, Testing, and Maintenance of Water Based Protection	The applicant will use this standard for modifications to the facility except as justified through documentation.
NFPA 30, 2003	Flammable and Combustible Liquids Code	Above ground storage tanks were installed using the provisions of this standard for guidance only. The applicant will use this standard for modifications to the facility except as justified through documentation.
NFPA 51B, 2003	Standards for Fire Prevention During Welding, Cutting, and Other Hotwork	None
NFPA 70, 2005	National Electric Code	The applicant will use this standard for modifications to the facility except as justified through documentation.
NFPA 72, 2002	National Fire Alarm Code	The applicant will use this standard for modifications to the facility except as justified through documentation.
NFPA 75, 2003	Standard for Protection of Electronic Computer/Data Processing Equipment	None
NFPA 80, 1999	Standard for Fire Doors and Fire Windows	The applicant will use this standard for modifications to the facility except as justified through documentation.
NFPA 101, 2003	Life Safety Code	None
NFPA 220, 1999	Standard on Types of Building Construction	The applicant will use this standard for modifications to the facility except as justified through documentation.

Code No. and Revision Year	Title	Exceptions
NFPA 232, 2000	Standard for the Protection of Records	There are several acceptable methods for storage of permanent records. If the NFPA 232 method of storage in 2-hour-rated containers is used, any exceptions to this standard will be documented and justified.
NFPA 241, 2000	Standard for Safeguarding Construction, Alteration, and Demolition Operations	None
NFPA 801, 2003	Standard for Fire Protection for Facilities Handling Radioactive Materials	The applicant will utilize this standard for any future modifications to the fire protection program as stated in Section 7.1.1 of the LA (USEC, 2006b).

The staff finds the use of these NFPA standards to be in accordance with the guidance of Section 7.4.3 of NUREG-1520 (NRC, 2002) in regard to nationally recognized codes and standards that may be used to measure reasonable assurance of fire safety. Therefore, the staff considers the use of the above NFPA standards to satisfy the requirements of 10 CFR 70.64(a) Baseline Design Criterion (3), "Fire Protection."

7.3.1 Fire Safety Management Measures

The applicant commits to implement configuration management, maintenance, training, procedures, audits and assessments, incident reporting and investigations, and record management as described in Chapter 11 of the LA (USEC, 2006b). These measures will ensure that fire protection IROFS are available and reliable. Management measures are evaluated in Chapter 11 of this SER.

The Fire Safety Manager is responsible for the Fire Safety program, including fire services, and reports to the Plant Support Manager. The Plant Support Manager has the authority to ensure that fire safety receives appropriate priority.

A Plant Safety Review Committee, as described in Chapter 2 of the LA (USEC, 2006b), will provide a review role of fire safety at the ACP. The membership, structure, and responsibilities of this multi disciplinary committee are defined in a plant procedure. The procedure includes the responsibility to review fire safety issues and to integrate changes to the plant with adequate consideration of fire safety.

The ACP Fire Safety program management measures are grouped into four areas:

- Fire prevention;
- Inspection, testing, and maintenance of fire protection systems;
- Emergency response organization (ERO) qualifications, drills, and training; and
- Pre-fire plans.

7.3.1.1 Fire Prevention

Fire prevention is defined in Section 7.1.1 of the LA (USEC, 2006b) as a program across the ACP to minimize the potential for an incipient fire. The following are major points the Fire Prevention program addresses:

- Workers are required to review and understand fire safety information;
- Building/facility inspections are conducted periodically and remedial actions are taken when conditions of concern are identified;
- General housekeeping practices and control of transient combustibles are established;
- Control of flammable and combustible liquids is handled in accordance with NFPA 30 (NFPA, 2003a);
- Ignition sources are controlled;
- Fire reports containing fire investigations and corrective actions are documented through the Corrective Action program;
- Smoking is restricted to designated areas; and
- Construction activities are performed in a manner that meets the requirements of NFPA 241 (NFPA, 2000b).

As part of the fire prevention program, there is also a program for control of impairments to Fire Protection systems. The Impairment Control program records and tracks impairments of the fire detection system, fire alarms, fire barriers, fire distribution system, and sprinkler system. The Impairment Control program may also result in the initiation of compensatory measures and the suspension of hot work or other hazardous activities.

Hot work control is also part of the Fire Prevention program. It is controlled by procedures complying with NFPA 51B (NFPA, 2003b) and applicable Occupational Safety and Health Administration (OSHA) requirements per 10 CFR Part 1910. The permit system ensures that cutting, welding, and other hot work conducted in plant areas not normally used for such purposes will be conducted using a permit system/process and performed in a manner that is consistent with industry fire prevention practices.

7.3.1.2 Inspection, Testing, and Maintenance of Fire Protection Systems

Fire protection equipment, such as sprinkler systems, fire alarms, and detection systems, is inspected and tested upon installation in accordance with NFPA 25 (NFPA, 2004). Periodic inspection and testing of fire protection equipment are performed by or overseen by trained personnel to help ensure that fire protection related IROFS are available and reliable. Testing and inspection of equipment is performed in accordance with procedures that include test frequencies as defined by the Fire Safety Manager.

7.3.1.3 ERO Qualifications, Drills, and Training

The applicant will rely on a qualified provider to perform emergency response to fire and other types of accident scenarios occurring at the ACP. This provider is presently an on-site fire department maintained by the United States Enrichment Corporation, a subsidiary of the applicant.

According to Section 7.1.3 of the LA (USEC, 2006b), employees receive initial and biennial fire safety training as part of the General Employee Training on emergency preparedness. This includes emergency reporting, building/facility evacuation, and fire extinguisher familiarization.

Firefighter training is equivalent to the State certified firefighter training curriculum. Emergency medical response personnel meet requirements for State certification as emergency medical technicians and are usually also firefighters.

Qualified instructors provide a range of classroom and hands-on training to maintain standards of performance for all response personnel. Training needs are reviewed annually and the training program modified to meet identified needs. Records of the training activities are maintained. Training is based on national standard emergency response methodology with plant specific training on issues unique to the plant. Drills are conducted as part of the plant emergency plan, which is evaluated in Chapter 8 of this SER.

7.3.1.4 Pre-Fire Plans

Pre-fire plans are or will be developed for the following areas: (a) X-3001 Process Building; (b) X-3002 Process Building; (c) X-3012 Process Support Building; (d) X-3346 Feed and Customer Building; (e) X-3346A Feed and Product Shipping and Receiving Building; (f) X-3356 Product and Tails Withdrawal Building; (g) X-7725 Recycle/Assembly Facility; (h) X-7726 Centrifuge Training and Test Facility; (i) X-7727H Interplant Transfer Corridor; and (j) the Cylinder Storage Yards (X-745G-2, X-745H, X-7746N, X-7746S, X-7746E, X-7746W, and X-7756S).

The pre-fire plans will contain the following information (as applicable):

- Facility description/construction;
- Specific hazards to emergency responders;
- Search and rescue considerations;
- Fire protection equipment/systems available;
- Utility shut-offs/start-ups;
- Fire loading concerns;
- Unique fire fighting strategy and tactics;
- Fire extension concerns; and
- Ventilation methodology.

The pre-fire plans will be reviewed as part of building inspections. As buildings are modified to meet changing operations, pre-fire plans will be reviewed and may be updated to assure that revised conditions are addressed.

7.3.1.5 Fire Safety Management Measures Conclusions

Based on its review, the staff reached the following conclusions:

Consistent with Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures reflect a commitment to ensure that the IROFS, as identified in the ISA Summary, are available and reliable, and to ensure that the facility maintains fire safety awareness among employees, controls transient ignition sources and combustibles, and maintains a readiness to extinguish or limit the consequences of fire. Therefore, these

measures are acceptable.

Consistent with Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures identify a senior level manager who has the authority and staff to ensure that fire safety receives appropriate priority, and are, therefore, acceptable.

Consistent with Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures identify a facility safety committee staffed by managers of different disciplines to integrate facility modifications, and are, therefore, acceptable.

Consistent with Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures include fire prevention; inspection, testing, and maintenance of fire protection systems; ERO qualifications, drills, and training; and pre-fire plans as recommended by NFPA 801 (NFPA, 2003e), and are, therefore, acceptable.

Consistent with Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures are documented in sufficient detail to identify their relationship to, and functions for normal operations; anticipated (off-normal) events; and accident safety (i.e., IROFS), and are, therefore, acceptable.

Based on the above conclusion, the staff has reasonable assurance that the applicant's fire safety management measures meet the requirements of 10 CFR 30.33, 40.32, 70.22, 70.61, 70.64, and 70.65 as they pertain to the fire protection aspects of the facility.

7.3.2 FHA

FHAs ensure that the fire prevention and fire protection requirements of buildings and fire areas have been evaluated and incorporated. The applicant has performed or will perform FHAs for the following buildings and areas: (a) X-3001 Process Building; (b) X-3002 Process Building; (c) X-3012 Process Support Building; (d) X-3346 Feed and Customer Building; (e) X-3346A Feed and Product Shipping and Receiving Building; (f) X-3356 Product and Tails Withdrawal Building; (g) X-7725 Recycle/Assembly Facility; (h) X-7726 Centrifuge Training and Test Facility; (i) X-7727H Interplant Transfer Corridor; and (j) the Cylinder Storage Yards (X-745G-2, X-745H, X-7746N, X-7746S, X-7746E, X-7746W, and X-7756S). In Section 7.2 of the LA (USEC, 2006b), the applicant states that an FHA will be performed for X-745H, a cylinder storage yard, before construction. Information developed from the FHAs was used by the applicant in the ISA to determine the credible fire accident scenarios, their likelihood of occurrence, the associated consequences, and the necessary control measures to reduce the likelihood of occurrence and/or the consequences to meet performance requirements.

The FHAs include the following elements:

- A listing of codes and standards used for the design of fire protection systems;
- Descriptions of characteristics associated with potential fires for areas that contain combustible materials;
- Listing of fire protection system criteria;
- Descriptions of performance criteria for active and passive fire protection systems;
- Descriptions of the design of automatic fire suppression systems;

- Definition of methods for fire prevention, fire extinguishing, fire control, and control of hazards created by fire;
- Identification of dangerous and hazardous combustibles and their quantities;
- Identification of the types of fires and severity, duration, and potential hazards created for each fire scenario reviewed;
- Definition of the essential electric circuit integrity needed for safe shutdown of safety related equipment (if required); and
- Evaluation of life safety, protection of critical process/safety equipment, lightning protection, provisions to limit contamination, potential for radioactive release, and restoration of the facility after a fire.

Building surveys are also conducted on an annual and/or semiannual basis to ensure that safety concerns are continually assessed and issues addressed as necessary to minimize the risk from fire related accidents.

Consistent with Section 7.4.3.2 of NUREG-1520 (NRC, 2002), an FHA was performed for each process area and each FHA included a description by fire area, of the fuel loading, fire scenarios, methods of consequence analysis, the potential consequences, and a description of mitigative controls, and is, therefore, acceptable. In addition, based on its review of the above information, the staff concludes with reasonable assurance that the applicant's performance of FHAs meets the requirements of 10 CFR 30.33, 40.32, 70.22, 70.61, 70.64, and 70.65, as they pertain to the fire protection aspects of the facility.

7.3.3 Facility Design in Regard to Fire Protection

7.3.3.1 Facility Passive-Engineered Fire Protection Systems

The construction of planned and existing primary buildings (X-3001, X-3002, X-3012, X-3346, X-3346A, X-3356, X-7725, X-7726, and X-7727H) consists of steel structural elements, concrete floors, corrugated metal walls, and a built-up roofing material on a metal deck. The buildings are considered to be non-combustible but do not have fire-rated perimeter walls, floors, or roof. This meets the requirements of NFPA 220 (NFPA, 1999b) for Type II 000 construction. This also meets the requirements of NFPA 801 (NFPA, 2003e). Also, fire resistant construction is provided for the support areas, the standby diesel generator room, the area control room and computer equipment areas, and exit stairways. Fire areas are considered to be any location bounded by fire rated construction with a minimum rating of 2 hours and equivalently rated doors, dampers, or penetration seals.

The applicant will limit the amount of fixed combustibles that could increase the severity of or may help to propagate a fire. As a result, a fire involving transient combustibles or a postulated large fire from combustible liquid would not spread by means of the facility construction materials. The applicant's ISA Summary (USEC, 2006a) credits the building construction (an initial condition) along with control of combustibles in certain postulated scenarios for limiting propagation and severity of a fire.

Fire areas are considered to be any location bounded by fire rated construction with a minimum rating of 2 hours and equivalent fire rated doors, dampers, or penetration seals. The X-3001, X-3002, X-3012, X-3346A, X-3356, and non-contiguous cylinder storage yards are each considered single fire areas. The X-7725 and X-7726 facilities and X-7727H corridor are

considered as a single fire area and the X-3346 building is divided into two fire areas. Based on the low combustible loading, distance between buildings, types of construction, and locations of hazards, the staff concludes that the applicant's designation of fire areas is adequate.

Electrical systems will be installed in accordance with NFPA 70, National Electric Code, (NFPA, 2005).

Life safety aspects of existing buildings have been evaluated using the Life Safety Code (NFPA, 2003d). Exit arrangements were found to be adequate based on the low occupancy levels, low combustible loading, large number of exits, and fixed fire suppression systems.

In that the existing fire services provider is an on-site fire department, security considerations to allow the fire responding trucks and firefighters quick access to the site are not required. Security provisions to maintain control of classified material during fire events are addressed in the Security program (USEC, 2006c) for the ACP.

Although the applicant has not listed the NFPA standard for Lightning Protection as applicable to its facility, the applicant states in Section 1.3.3.3 of the LA (USEC, 2006b) that the facility buildings are well grounded and some have installed lightning protection. Also, the applicant has evaluated lightning strikes in its ISA Summary (USEC, 2006a) and has determined that additional controls are not required. The staff concludes that compliance with the NFPA standard for Lightning Protection is not required in order for the facility to comply with the performance requirements of 10 CFR 70.61, nor the baseline design criteria of 10 CFR 70.64.

7.3.3.2 Facility Active-Engineered Fire Protection Systems

Fire Alarm System: The fire alarm system will consist of water flow alarms connected to the fire alarm system and manual pull stations located in the primary facility buildings (X-3001, X-3002, X-3012, X-3346, X-3346A, X-3356, X-7725, X-7726, and X-7727H). The audible indication of a fire is provided by means of a separate plant public announcement system. The key safety function of the fire alarm system is to provide automatic detection and the means of automatic (water flow alarms) and manual initiation of a plant emergency response. As a result, the fire alarm system and components are important to meeting the safety performance requirements of initiating a response by the plant emergency response to suppress a fire before conditions increase the risk of a release of uranium hexafluoride (UF_6). The building fire alarm system will meet the requirements of NFPA 72 (NFPA, 2002c).

Engineered Automatic Fire Suppression Systems: The applicant will provide sprinkler protection throughout the ACP to minimize the risk of fire. There are no water exclusion areas in the ACP in regard to nuclear criticality safety. However, the applicant identified criticality concerns in the X-3346 Customer Service Area such that floors are required to have no diking or areas where ponding can occur. The sprinkler systems are identified as protection that minimizes the risk of fires that could affect the safety of nuclear operations. The applicant stated that the existing automatic sprinkler system was designed to exceed the NFPA 13 (NFPA, 2002b) recommended sprinkler density for Ordinary Hazard Group 1 occupancies. New suppression systems will meet NFPA 13 (NFPA, 2002b) and NFPA 15 (NFPA, 2001) standards to ensure availability and reliability. The PORTS high-pressure fire water system provides the water supply needed for automatic fire suppression.

Firewater Distribution System: The ACP fire suppression systems are part of the U.S. Department of Energy (DOE) reservation firewater distribution system. The firewater distribution system is maintained by United States Enrichment Corporation, a subsidiary of the applicant. The firewater distribution system is designed such that each ACP building can be supplied by at least two sources of water. The firewater storage tanks include one 300,000 gallon elevated tank and two 2,000,000 gallon surface tanks. The firewater pumps include two electric pumps and one diesel pump each with a capacity to pump 4,000 gallons per minute.

Firewater runoff to the environment is controlled by the presence of holding ponds that can reduce or terminate releases as necessary to minimize environmental impact.

7.3.3.3 Mobile and Portable Equipment

A qualified supplier provides mobile and portable fire protection equipment. Portable fire extinguishers are available throughout the ACP. Size, selection, and distribution of fire extinguishers are determined in accordance with NFPA 10 (NFPA, 2002a).

7.3.3.4 Conclusion on Facility Design Regarding Fire Protection

The applicant has addressed building construction, fire area determination, electrical installation, life safety, drainage, and lightning protection adequately in the application. A description of ventilation characteristics as they relate to fire protection and fire hazards will be provided in the FHAs. The applicant has also adequately addressed criticality safety, environmental concerns, and physical security to the extent applicable to the facility. As stated in the introduction to Section 7.3 of this SER, the applicant meets the baseline design criteria through compliance with accepted consensus standards. The applicant meets defense-in-depth requirements as evaluated in Appendix D of this SER. Consistent with Section 7.4.3.3 of NUREG-1520 (NRC, 2002), the application documents the fire safety considerations used in the general design of the facilities containing licensed material, or facilities that impose an exposure threat to radiological facilities, and is, therefore, acceptable. In addition, the applicant's facility fire protection features meet the requirements of 10 CFR 30.33, 40.32, 70.22, 70.61, 70.64, and 70.65 as they pertain to the fire protection aspects of the facility.

7.3.4 Process Fire Safety

Process fire safety is discussed in the ACP ISA Summary (USEC, 2006a). The ISA Summary (USEC, 2006a) contains:

- Accident analysis including major fire scenarios;
- The effects of fire safety measures in preventing fire scenarios;
- The effect of the fire protection system in controlling and mitigating the fire scenarios; and
- Toxic and radiological hazards from fire related scenarios.

The staff review of process fire hazards and special hazards, fire related accident scenarios, and fire related IROFS are addressed in Appendix D, "Fire Safety," of this SER.

7.3.5 Fire Safety and Emergency Response

Manual fire suppression capability will be provided by a fire service department which is manned continuously that is adjacent to the DOE reservation. The fire service department will have a minimum staff of four fire fighters and one supervisor per shift. A minimum of four fire fighters will be required for entry and backup. Equipment requirements will include one pumper truck with a minimum capacity of 1,000 gpm, one ambulance, and one hazardous material truck with radiological and rescue equipment. The fire department will provide personnel who are trained to meet or exceed state requirements for fire fighting. Because the fire department will be a professional department and not an industrial fire brigade, and based on the fire department description in Section 7.1.3 of the LA (USEC, 2006b), the staff concludes that compliance with applicable parts of NFPA 600, "Standards on Industrial Fire Brigades" (NFPA, 2000c), will be met or exceeded. The time to apply water on a fire will not exceed 20 minutes, 90 percent of the time.

The adequacy of fire protection IROFS and the fire protection provided for areas where licensed material is present is evaluated in Appendix D of this SER.

The applicant's use of fire protection codes and standards is addressed in Section 7.3 of this SER.

Consistent with Section 7.4.3.5 of NUREG-1520 (NRC, 2002), the LA (USEC, 2006b) documents the fire protection systems and fire emergency response organizations that are provided, and is, therefore, acceptable. In addition, the applicant's emergency response capability meets the requirements of 10 CFR 30.33, 40.32, 70.22, 70.61, 70.64, and 70.65, as they pertain to the fire protection aspects of the facility.

7.4 EVALUATION FINDINGS

The applicant has established a fire protection function meeting the acceptance criteria in Chapter 7 of the "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (2002). The function includes a Facility Safety Review Committee responsible for integrating modifications to the facility and a Fire safety Manager responsible for day-to-day program implementation. Fire prevention, inspection, testing, and maintenance of fire protection systems, and the qualification, drills, and training of facility personnel are in accordance with applicable NFPA codes and standards.

The applicant has conducted risk analyses in accordance with NFPA 801, "Standards for Fire Protection for Facilities Handling Radioactive Material." The FHAs identified credible fire scenarios that bound the fire risk. The ISA used these scenarios and identified fire protection IROFS. Procedures are in place (through training and pre-fire plans) to allow the fire department efficient access to process areas during fire emergencies.

The applicant has demonstrated that appropriate fire safety considerations were incorporated in the design of its facilities. The applicant has also demonstrated that the facility has appropriate active fire protection systems.

The applicant has demonstrated the availability of an adequate fire response organization.

The staff concludes that the applicant's submittals provide sufficient information in accordance with requirements of 10 CFR 30.33, 40.32, 70.22, and 70.65 regarding potential fire hazards, consequences, and required controls for the proposed ACP processes. NRC staff determined that the applicant demonstrated compliance with the performance requirements of 10 CFR 70.61 for fire protection related to postulated accident scenarios. The design the applicant proposes also satisfies the requirements of 10 CFR 70.64(a) Baseline Design Criterion (3) "Fire Protection" as well as 10 CFR 70.64 (b), defense in depth.

7.5 REFERENCES

(IEEE, 2002) Institute of Electrical and Electronics Engineers (IEEE). IEEE Standard 484, "IEEE Recommended Practice for Installation Design and Implementation of Vented Lead-Acid Batteries for Stationary Applications," 2002.

(NFPA, 1999a) National Fire Protection Association (NFPA). NFPA 80, "Standard for Fire Doors and Fire Windows," 1999.

(NFPA, 1999b) National Fire Protection Association (NFPA). NFPA 220, "Standard on Types of Building Construction," 1999.

(NFPA, 2000a) National Fire Protection Association (NFPA). NFPA 232, "Standard for Protection of Records," 2000.

(NFPA, 2000b) National Fire Protection Association (NFPA). NFPA 241, "Standards for Safeguarding Construction, Alteration, and Demolition Operations," 2000.

(NFPA, 2000c) National Fire Protection Association (NFPA). NFPA 600, "Standards on Industrial Fire Brigades," 2000.

(NFPA, 2001) National Fire Protection Association (NFPA). NFPA 15, "Standard for Waterspray Fixed System for Fire Protection," 2001.

(NFPA, 2002a) National Fire Protection Association (NFPA). NFPA 10, "Standard for Portable Fire Extinguishers," 2002.

(NFPA, 2002b) National Fire Protection Association (NFPA). NFPA 13, "Standards for the Installation of Sprinkler Systems," 2002.

(NFPA, 2002c) National Fire Protection Association (NFPA). NFPA 72, "National Fire Alarm Code," 2002.

(NFPA, 2003a) National Fire Protection Association (NFPA). NFPA 30, "Flammable and Combustible Liquids Code," 2003.

(NFPA, 2003b) National Fire Protection Association (NFPA). NFPA 51B, "Standard for Fire Prevention During Welding, Cutting, and Other Hot Work," 2003.

(NFPA, 2003c) National Fire Protection Association (NFPA). NFPA 75, "Standard for Protection of Electronic Computer/Data Processing Equipment," 2003.

(NFPA, 2003d) National Fire Protection Association (NFPA). NFPA 101, “Life Safety Code,” 2003.

(NFPA, 2003e) National Fire Protection Association (NFPA). NFPA 801, “Standard for Fire Protection for Facilities handling Radioactive Materials,” 2003.

(NFPA, 2004) National Fire Protection Association, (NFPA). NFPA 25, “Standard for the Inspection, Testing, and Maintenance of Water Based Protection,” 2004.

(NFPA, 2005) National Fire Protection Association (NFPA). NFPA 70, “National Electric Code,” 2005.

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

(USEC, 2006a) USEC Inc. (USEC). “Integrated Safety Analysis Summary for the American Centrifuge Plant in Piketon, Ohio,” Revision 14, August 2006.

(USEC, 2006b) USEC Inc. (USEC). “License Application for the American Centrifuge Plant in Piketon, Ohio,” Revision 18, September 2006.

(USEC, 2006c) USEC Inc. (USEC). “Security Program for the American Centrifuge Plant in Piketon, Ohio,” Revision 5, August 2006.

8.0 EMERGENCY MANAGEMENT

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's emergency management plan is to evaluate whether the application provides, 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

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 NRC. If an applicant is required to submit an Emergency Plan (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 10 CFR 30.32(i)(3), 10 CFR 40.31(j)(3), and 10 CFR 70.22(i)(3) contain the information that must be included in the EP. 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.

8.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the emergency management program of the license application (LA) (USEC, 2006b) is contained in Chapter 8 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). Section 8.3.2 of NUREG-1520 (NRC, 2002) lists the areas of review for an application that demonstrates that an EP is not required by including an evaluation or references an Integrated Safety Analysis (ISA). Section 8.6.2.2 of NUREG-1520 (NRC, 2002) outlines the review procedures to be used if such an application is submitted. Sections 8.3.2 and 8.6.2.2 of NUREG-1520 (NRC, 2002) are not applicable to the LA (USEC, 2006b) since the applicant submitted an EP (USEC, 2006a). The remainder of Chapter 8 is applicable to the American Centrifuge Plant (ACP) facility. The acceptance criteria applicable to this review are contained in Section 8.4.3.1 of NUREG-1520 (NRC, 2002). In addition, the staff used Regulatory Guide 3.67, (NRC, 1992), which is referenced in NUREG-1520 (NRC, 2002) Section 8.4.2. The staff also used Regulatory Guide 3.67 (NRC, 1992), which describes the information in an EP for all fuel cycle and materials facilities to comply with the regulatory requirements in 10 CFR 70.22(i)(3) and 70.64(a)(6).

8.3 STAFF REVIEW AND ANALYSIS

The applicant submitted an EP (USEC, 2006a) to encompass the ACP and other on-going activities on the U.S. Department of Energy (DOE) reservation in Pike County, Ohio. The submitted EP (USEC, 2006a) is based on the Portsmouth Gaseous Diffusion Plant (PORTS) EP, which encompasses only current activities at the site (USEC, 2005). As part of certifying PORTS under 10 CFR Part 76, NRC had reviewed and approved the PORTS EP (USEC, 2005). The requirements of 10 CFR 76.91 are the same as the requirements of 10 CFR

30.32(i)(3), 10 CFR 40.31(j)(3), and 10 CFR 70.22(i)(3), which are applicable to the ACP. NRC staff reviewed the following elements against the EP (USEC, 2006a), as they relate to the proposed ACP, along with the information provided by the applicant in the LA (USEC, 2006b).

8.3.1 Facility Description

Section 1.0 of the EP (USEC, 2006a) describes the ACP site and the area near the site. The information provided includes the location and size of the site, site characteristics, on-site activities, and rail and roadways traversing the site. Information is also provided on land use adjacent to the reservation, population centers, and facilities located near the site.

Appendix A of the EP (USEC, 2006a) describes licensed activities to be performed at the proposed ACP, Appendix B of the EP (USEC, 2006a) describes the certified activities performed at the Gaseous Diffusion Plant (GDP), and Appendix C of the EP (USEC, 2006a) describes the licensed activities performed at the American Centrifuge Lead Cascade plant. Theses appendices contain the following information:

- A description of the enrichment process;
- A list of hazardous chemicals to be used at the facility, including location and typical quantity used;
- A description of the plant;
- A summary of the facility's monitored stacks and vents, including locations, stack heights, flow rates, and emission control device efficiencies; and
- A listing of possession limits of licensed materials.

The staff has reviewed the information and description contained in appendices A through C and finds that it is sufficient for emergency planning purposes.

Maps and drawings of the reservation and surrounding area are also included in Section 1 of the EP (USEC, 2006a). The maps and drawings include: (a) on-site and near off-site structures; (b) building numbers and labels; (c) roads and parking lots on-site and main roads near the site; (d) site boundaries showing fences and gates; (e) major features; and (f) appropriate bodies of water.

Section 10.0 of the EP (USEC, 2006a) states that the applicant complies with the Emergency Planning and Community Right to Know Act in accordance with 10 CFR 30.32(i)(3)(xiii), 10 CFR 40.31(j)(3)(xiii), and 10 CFR 70.22(i)(3)(xiii). The EP (USEC, 2006a) and appropriate plant procedures would be used in case of a hazardous chemical release emergency. There are also plant procedures for releases that are not considered emergencies. Procedures describe actions to be implemented on a hazardous chemical release emergency, from evacuating the area of release or spill to terminating the incident and proceeding to the recovery phase.

The staff reviewed the applicant's facility description against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 8.4.3.1.1 and, based on its review, the staff finds that:

1. The applicant has provided detailed maps and drawings of the site which showed;

    a. On-site and near off-site structures, building numbers and labels;
    b. Roads and parking lots on-site and main roads near the site;

c. Site boundaries showing fences and gates;

d. Major features; and

e. Appropriate bodies of water;

2. The applicant has provided a general area map;

3. The applicant has provided information on stack heights, typical stack flow rates, and efficiencies of any emission-control devices;

4. The applicant has provided a general description of licensed and other major activities conducted at the facility, and type, form, and quantities of radioactive and other hazardous materials that are normally on-site by location and hazardous characteristics that are important to emergency management in Appendix A of the EP (USEC, 2006a); and

5. The applicant has certified that it has met all responsibilities under the Emergency Planning and Community Right to Know Act of 1986, Title III, Public Law 99-499, in accordance with 10 CFR 70.22(i)(3)(xiii).

8.3.2 On-site and Off-site Emergency Facilities

Section 6.0 of the EP (USEC, 2006a) describes the on-site emergency facilities available during an emergency. The X-1020 building will be the primary emergency operations center (EOC) responsible for coordinating on-site response and mitigation, as well as off-site interface activities. EOC personnel will be responsible for conducting technical interactions with federal, state, and local officials; generating emergency information for public information activities; ensuring required support to the incident scene; and coordinating support for on-site response and mitigation. The X-300 facility, which will be the GDP Plant Control Facility, will be the alternate EOC if the primary EOC becomes uninhabitable. A mobile communications vehicle will also be available as an EOC, if necessary.

Other on-site emergency facilities include decontamination facilities, the Joint Public Information Center, and medical facilities. Section 4.3 of the EP (USEC, 2006a) and Appendix D of the EP (USEC, 2006a) describe and list off-site emergency support by name and location for medical assistance, fire control, evacuation, ambulance services, and law enforcement assistance.

In Section 6.4 of the EP (USEC, 2006a), the applicant provides a description of emergency monitoring equipment and has committed to maintain sufficient and appropriate equipment and supplies to support emergency response activities, including: (a) radiation detection equipment; (b) emergency vehicles; (c) monitoring stations; (d) instrumentation to measure chemical toxic material releases; (e) respiratory protection equipment; (f) firefighting equipment; and (g) protective clothing. The equipment and supplies will be inspected, inventoried, and operationally tested quarterly and after each use. A discussion and list of the type of radiation protection and monitoring instruments, personnel monitoring, and calibration procedures for this facility are contained in Section 4.3.7 of this Safety Evaluation Report (SER).

Section 6.2 of the EP (USEC, 2006a) describes the available primary and backup communications systems for emergency needs. This includes telephone systems, the public address system, radio systems, and pager systems. A public warning system will also be available to provide emergency notification to the public using warning sirens and emergency alert system announcements. The system's sirens will be tested monthly, and audible testing will be performed semiannually. The telephone systems serve as the primary emergency communication system.

The staff reviewed the applicant's on-site and off-site emergency facilities against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 8.4.3.1.2 and, based on its review, the staff finds that:

1. The applicant has listed and described on-site and off-site facilities by location and purpose;

2. The applicant has provided a description of emergency monitoring equipment that will be available for personnel and area monitoring and for assessing releases of radioactive material or hazardous chemicals to the environment;

3. The applicant has provided a description of the on-site and off-site services that support emergency response operation including:

    a. Decontamination facilities;

    b. Medical treatment facilities;

    c. First aid personnel;

    d. Fire fighters;

    e. Law enforcement assistance; and

    f. Ambulance services;

4. The applicant has committed to having:

    a. Facilities of adequate size with a stated purpose and appropriate location;

    b. Adequate backup facilities required by the EP (USEC, 2006a) and supporting documents that are available and ready for use;

    c. Appropriate equipment and supplies necessary to support emergency response activities that are accessible during accident conditions;

    d. Emergency equipment that will be inventoried, tested, and serviced on a periodic basis to ensure accountability and reliability;

    e. Sufficient reliable primary and backup communications channels that are available to accommodate emergency needs;

f. Off-site emergency resources and services that are identified and ready to ensure their timely mobilization and use;

g. Operational engineering information, such as current as-built drawings and procedures, that are readily available in the emergency facilities;

h. Sufficient equipment for personnel protection and monitoring; and

i. Systems in place to alert on-site and off-site personnel in case of an emergency.

8.3.3 Types of Accidents

Section 2.0 of the EP (USEC, 2006a) describes the potential accidents that may require protective actions. Also, Table A-4 of Appendix A of the EP (USEC, 2006a) provides a summary of event scenarios by processes and location, including complicating factors and possible on-site and off-site consequences. Table A-4 also provides the unmitigated off-site consequence levels (high, intermediate, or low) and the ISA event number so as to obtain dose projections from the ISA, if required. The events include fires, explosions, loss of confinement, direct radiological/chemical exposure events, nuclear criticality, external events, and natural phenomenon events. The most extreme credible scenario at the ACP was determined to be an accident involving a large release of uranium hexafluoride (UF_6). Events have been identified and analyzed to assess the potential consequences to plant workers, the public, the environment, and on-site and off-site property. Events at the ACP with potential unmitigated off-site consequences have been identified in Table A-4 of the EP (USEC, 2006a). Items relied on for safety (IROFS) will be required for these events to reduce the likelihood and/or consequences to meet the performance requirements of 10 CFR 70.61. The applicant has indicated it will use ANSI/ANS-8.23-1997 (ANSI/ANS, 1997) as guidance for criticality accident emergency planning. For references to this standard, see Section 5.4.4 of the LA (USEC, 2006b) and Section 2.2.4 of the EP (USEC, 2006a).

The applicant also discusses potential events by processes and location that may occur at the GDP and Lead Cascade in Appendices B and C of the EP (USEC, 2006a). The applicant will make changes to the EP (USEC, 2006a), if necessary, when the details of the design and safety analysis of the depleted uranium hexafluoride facility are available.

The staff reviewed the applicant's description of types of accidents against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 8.4.3.1.3 and, based on its review, the staff finds that for each general type of accident identified in the ISA, for which protective actions may be needed, the EP (USEC, 2006a) has described the following:

1. The process and physical location(s) where the accidents could occur;

2. Complicating factors and possible on-site and off-site consequences, including releases of non-radioactive hazardous chemicals incident to the processing of licensed material that could impact emergency response efforts;

3. The accident sequence that has the potential for the greatest radiological and/or toxic chemical impact; and

4. ISA event numbers where projections of doses and toxic substance concentrations as a function of distance and time for various meteorological stability classes may be obtained.

8.3.4 Classification of Accidents

Section 3.0 of the EP (USEC, 2006a) describes the system used to classify an emergency as either an Alert or a Site Area Emergency (SAE), as required by 10 CFR 70.22(i)(3)(iii) and defines each classification level. Emergency Action Levels (EALs) have been established to determine if an event meets the criteria of an Alert or SAE. The EALs are based on the U.S. Environmental Protection Agency's (EPA) Protective Action Guidelines (PAGs) (EPA, 1992). This method of classifying events is acceptable to the staff as explained in NUREG-1520 (NRC, 2002), Section 8.4.3.1.4. Tables 3.2-1 and 3.2-2 in the EP (USEC, 2006a) contain the EAL criteria for potential events. Table 3.2-1 provides the EALs and consequences associated with Alerts, and Table 3.2-2 provides the EALs associated with SAEs. The Incident Commander will be responsible for classifying events and activating the Emergency Response Organization (ERO). Section 4.2.1 of the EP (USEC, 2006a) indicates the succession for the Incident Commander position. The Incident Commander becomes the Crisis Manager once the Emergency Operations Center becomes operational. The Crisis Manger then assumes the responsibility for declaring the appropriate class of emergency and makes any changes to the emergency classification, including event termination.

The staff reviewed the applicant's classification of accidents against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.4, and, based on its review, the staff finds that the applicant:

1. Classifies accidents as Alert or Site Area Emergency;

2. Identifies the classification expected for each accident type;

3. Specifies EALs, at which an alert or SAE will be declared, and are based on the EPA's PAGs (EPA, 1992); and

4. Designates the personnel positions and alternates for accident classification during normal operations and back shifts.

8.3.5 Detection of Accidents

Section 2.2 of the EP (USEC, 2006a) discusses the methods and systems used for detecting accidents for both radioactive materials and toxic chemical releases at the site, including:

- UF_6 detection equipment and associated alarms;
- Fire alarms;
- Radiation monitors;
- Automatic sprinkler systems;
- Chemical detectors;
- Criticality accident alarm system; and
- Visual observation of UF_6 release through sight or smell.

The ACP, GDP, and Lead Cascade have Area Control Rooms where process equipment will be monitored, alarms will be received to alert the operating staff, and corrective actions can be performed to mitigate potential consequences of accidents. As discussed in Section 5.0 of the EP (USEC, 2006a), emergency procedures have been established to direct the operating staff's response during anticipated accidents.

The staff reviewed the applicant's detection of accidents against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.5, and, based on its review, the staff finds that the EP (USEC, 2006a) identifies:

1. The means of detecting the accident;

2. The means of detecting any release of radioactive material or hazardous chemicals incident to the processing of licensed materials;

3. The means of alerting the operating staff; and

4. The anticipated response of the operating staff.

8.3.6 Mitigation of Consequences

Sections 5.3 and 5.4 of the EP (USEC, 2006a) discuss the methods used to mitigate consequences of potential accidents to both on-site and off-site personnel. Protective actions on-site include alerting personnel, assembling and accounting for personnel, sheltering in place, evacuation, monitoring, and decontaminating.

The Crisis Manager will provide recommended off-site protective actions to local officials. These actions may include sheltering in place, evacuation, or issuing advisories stating that no action will be necessary.

In Section 7.6 of the EP (USEC, 2006a), the applicant discusses the maintenance and inventory of emergency equipment, instrumentation, and supplies. The equipment includes protective clothing, fire fighting equipment, sampling equipment, respiratory protection equipment, emergency air supplies, materials for radiological and toxicological monitoring, and other equipment. Quarterly inventories, inspections, and operational checks of this equipment will be performed and any deficiencies will be corrected.

The staff reviewed the applicant's mitigation of consequences against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.6, and, based on its review, the staff finds that the EP (USEC, 2006a) describes measures and equipment to be used for safe shutdown and mitigating the consequences to workers on-site and off-site, as well as to the public off-site for each accident.

8.3.7 Assessment of Releases

In Section 5.2 of the EP (USEC, 2006a), the applicant describes the methods that will be used to assess releases both on-site and off-site during an event. The methods include: (a) increasing surveillance of applicable instrumentation and visual observation of conditions; (b) determining resources necessary to mitigate the event; (c) monitoring conditions for potential

changes in classification level; (d) assessing on-site and off-site exposures; (e) determining the need for sheltering or evacuation both on-site; and (f) off-site and communicating necessary information to off-site officials. The information communicated to off-site officials during off-site releases includes:

- Specific material information;
- Release information;
- Plume direction;
- Projected plume location;
- Meteorological information; and
- Field monitoring results.

Projected movement and dispersion of chemical release plumes will be determined using the Areal Locations of Hazardous Atmospheres (ALOHA) computer program. The ALOHA code is a well known code for this purpose and acceptable to the staff. Post-accident assessment will include individual monitoring and sampling of water, air, and soil.

The staff reviewed the applicant's assessment of releases against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.7, and, based on its review, the staff finds that the EP (USEC, 2006a) describes:

1. Procedures for assessing the release of radioactive material or hazardous chemicals incident to the processing of licensed material, including:

    a. Procedures for estimating or measuring the release rate or source term;

    b. Valid computer codes used to project dose or concentration of material;

    c. Details of on-site and off-site sampling and monitoring to be used; and

    d. Method for assessing collateral damage to the facility.

2. That there is no need to provide procedures for validating codes used to assess releases of radioactive material or hazardous chemicals since the applicant is using the well known and accepted ALOHA code.

8.3.8 Responsibilities

Section 4.0 of the EP (USEC, 2006a) discusses the responsibilities of the ERO. Section 4.2 of the EP (USEC, 2006a) states that the applicant will have sufficient staffing and resources. Figure 4-2 of the EP (USEC, 2006a) illustrates the ERO and interfaces with DOE, GDP, and Uranium Disposition Services. Section 4.2.2.2 of the EP (USEC, 2006a) provides a brief description of each ERO position. The ACP Plant Support Manager will be responsible for maintaining the EP (USEC, 2006a) for the site. The GDP Plant Shift Superintendent (PSS) serves as the Incident Commander for the site. The Incident Commander will be responsible for making proper notifications of abnormal conditions, determining the severity of the event, declaring an emergency, and initiating appropriate response. Responsibilities of the ERO include:

- Event categorization;
- Notification;
- Protective action recommendations;
- Management and decision making;
- Control of on-site emergency activities;
- Consequence assessment;
- Emergency public information;
- Activation and coordination of on-site response resources, security, communications, and administrative support;
- Coordination and liaison with off-site support and response organizations; and
- Downgrading and/or terminating of emergencies.

If an emergency condition, the GDP occurs, the PSS will be notified. The PSS will assume the Incident Commander duties in the event of a declared emergency. The PSS will work with the ACP Shift Operations Manager to determine the appropriate emergency response for ACP emergencies. The Director, American Centrifuge Plant, or designee, will subsequently become the Crisis Manager and is authorized to declare an emergency, initiate the appropriate response, and assign a Recovery Manager when emergency conditions no longer exist.

As discussed in Section 4.3 of the EP (USEC, 2006a), letters of agreement have been entered into with local off-site agencies including services for medical assistance, fire control, law enforcement, and ambulance services.

The staff reviewed the applicant's responsibilities against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.8, and, based on its review, the staff finds that the EP (USEC, 2006a):

1. Describes the organizational structure and chain of command;

2. Contains a commitment that staffing and resources will be sufficient to accomplish all assigned tasks;

3. Describes responsibilities and authority for each management, supervisory, and professional position;

4. Describes the interfaces with supporting groups, both on-site and off-site;

5. Contains a commitment to having mutual cooperation agreements with local agencies, such as fire, police, ambulance/rescue, and medical units;

6. Contains a commitment to having plant management measures that will audit and assess emergency preparedness to ensure site readiness to handle emergencies and to identify and correct problems;

7. Describes the on-site ERO that will provide effective command and control of the site during the assessment, mitigation, and recovery phases of an accident;

8. Describes the emergency public information system that will provide advance and ongoing information to the media and public on subjects that would be discussed during

an emergency, such as radiation hazards, chemical hazards, site operation, and site EP; and

9. Describes the schedule of emergency preparedness procedure development that will provide for availability of procedures to support startup and operation of new processes/facilities on-site.

8.3.9 Notification and Coordination

Section 3.0 of the EP (USEC, 2006a) discusses the use of EALs to determine the classification of accidents. The applicant will also use pre-approved notification procedures and messages to minimize distraction of site personnel. The messages will include concise, preformatted information and follow-up messages to off-site authorities. As discussed above, Section 3.3 of the EP (USEC, 2006a) provides a commitment to notify off-site authorities, including NRC, within 1 hour of declaring an alert or SAE. This notification will be performed when a change in classification level occurs, or when protective action recommendations off-site will be required. Information to be included in these notifications is discussed in Section 8.3.10 of this SER.

Section 4.3 of the EP (USEC, 2006a) discusses the procedures for requesting off-site assistance when needed. The types of assistance available will be medical support, fire support, law enforcement assistance, and also support through Local, State, and Federal government agencies. When off-site support will be needed, the applicable off-site authorities will be notified by telephone or radio transmission.

The Joint Public Information Center provides timely information dissemination to the media and public during Site Area Emergencies and any other events that may generate interest from the media.

The Incident Commander will be responsible for classifying events and activating the ERO. The Incident Commander, or Crisis Manager, will also be responsible for the decisions regarding on-site protective actions to initiate, off-site protective actions to recommend, support from off-site organizations, and terminating the emergency and entering recovery mode.

Sections 5.6 and 5.7 of the EP (USEC, 2006a) discuss medical transportation and treatment of contaminated injured workers. Decontamination of injured personnel will be performed before transport only if medical conditions permit.

The staff reviewed the applicant's notification and coordination against the acceptance criteria in NUREG-1520 (NRC, 2002) Section 8.4.3.1.9 and, based on its review, the staff finds that:

1. The EP (USEC, 2006a) provides reasonable assurance that:

   a. Emergency events will be classified on the basis of the EP (USEC, 2006a);

   b. Notification procedures will minimize distraction of shift site personnel and will include concise, preformatted messages and follow-up messages to off-site authorities;

c. Information on the nature and magnitude of the hazard will be made available to appropriate emergency response personnel;

d. Radiological and chemical source terms will be available to appropriate on-site and off-site personnel;

e. When available, off-site field monitoring data will be used in the protective action recommendation process;

f. Protective Action Guidelines will be available and appropriate personnel will use them in a timely manner;

g. The emergency public information program will ensure timely dissemination of understandable information;

h. Systems will be in place, to alert, notify, and mobilize on-site and off-site response personnel in case of an emergency; and

i. Procedures will be in place to notify and coordinate with responsible parties when some personnel, equipment, and facility components are not available.

2. The EP (USEC, 2006a) describes how and by whom the following actions will be promptly and effectively taken:

a. Decision to declare an alert or site area emergency;

b. Activation of the on-site ERO during all shifts;

c. Prompt notification of off-site response authorities that an alert or site area emergency has been declared;

d. Notification to NRC operations center;

e. Decision regarding what on-site protective actions to initiate;

f. Decision regarding what off-site protective actions to recommend;

g. Decision to request support from off-site organizations; and

h. Decision to terminate the emergency or enter recovery mode.

8.3.10 Information to Be Communicated

Section 3.3 of the EP (USEC, 2006a) provides a description of the information to be communicated to off-site response organizations and NRC during emergency notifications. The applicant's staff will use a standard reporting checklist to facilitate timely notification to off-site authorities. Communication of information will be performed using forms and procedures developed for emergency notifications, including preplanned protective action recommendations.

The information communicated includes plant status conditions, radiological and other hazardous materials release data, recommendations for protective actions for off-site response organizations, and other applicable emergency information. In the event of an SAE or other events that may generate significant interest from the media, a Joint Public Information Center will be activated to provide timely information to the media and the public.

The applicant lists the off-site agencies that will be notified after an initial emergency declaration and lists off-site agencies that may be notified based on the nature and status of the incident. The applicant states that off-site protective actions will be the responsibility of government authorities. Section 5.4.2 of the EP (USEC, 2006a) discusses off-site protective action recommendations.

The staff reviewed the applicant's information to be communicated during an emergency against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.10, and, based on its review, the staff finds that the EP (USEC, 2006a) includes the following:

1. A standard reporting checklist to facilitate timely notification;

2. The types of information to be provided concerning facility status, radioactive releases or hazardous chemicals and protective action recommendations;

3. A commitment to have preplanned protective action recommendations made to each appropriate off-site organization;

4. The off-site officials to be notified, as a function of the classification of the event; and

5. The recommended actions to be implemented by off-site organizations for each accident treated in the EP (USEC, 2006a).

8.3.11 Training

Section 7.2 of the EP (USEC, 2006a) describes the types of training that general plant personnel, ERO and support personnel, other DOE reservation personnel, and off-site emergency support organizations receive. Training records will be retained to document readiness assurance. Formal training programs have been developed and updated, including the General EP Training (for on-site personnel who will be not members of the ERO), Specialized EP Training for the ERO, and Off-site Emergency Management Training. Sections 7.2.1, 7.2.2, and 7.2.3 of the EP (USEC, 2006a) describe the topics and general content of training programs used for training the licensee's on-site and off-site emergency response personnel. These training courses will be provided on a biennial basis.

Training requirements for each position will be outlined in the Emergency Management Training procedure, including frequency of training. The Training Manager will be responsible for the analysis, design, development, implementation, and evaluation of training programs. Training for use of protective equipment will be provided during both fire-fighting and hazardous material emergency response initial and refresher training.

The Off-site Emergency Management Training for non-applicant responders, such as fire, police, and medical emergency personnel, will include information on specific on-site hazards

and protective actions, emergency response actions of on-site personnel, and orientation tours.

The staff reviewed the applicant's training against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.11, and, based on its review, the staff finds that the EP (USEC, 2006a) includes a description of the frequency, performance objectives, and plans for the training that the applicant will provide workers on how to respond to an emergency, including:

1. The topics and general content of training programs;

2. The administration of the training program, including responsibility for training, the positions to be trained, the schedules of training, the frequency of retraining, use of team training, and the estimated number of hours of initial training and retraining;

3. The training to be provided on the use of protective equipment;

4. The training program for on-site personnel who are not members of the emergency response staff; and

5. Special instructions and orientation tours provided to non-applicant responders such as fire, police, and medical emergency personnel.

8.3.12 Safe Shutdown (Recovery and Facility Restoration)

Section 5.3.2 of the EP (USEC, 2006a) discusses the safe shutdown of process equipment or systems or isolation of operating systems, if necessary, because of emergency conditions. Section 9.0 of the EP (USEC, 2006a) describes the means of restoring the facility to a safe condition after an accident. The Crisis Manager will determine when the recovery phase of the emergency can be initiated based on the following criteria:

- Emergency conditions no longer meet any EAL;
- Affected building/facility or area will be in a stable condition and can be maintained in that condition;
- Fire or other similar emergency conditions no longer constitute a hazard;
- Releases of hazardous materials to the environment have ceased or will be controlled; and
- Discussions with the ERO and appropriate off-site agencies identify no valid reason to continue in any emergency classification.

Recovery and restoration activities will be conducted such that exposures will be maintained in accordance with the as low as is reasonably achievable principle. A recovery organization will be established with key operating and management positions as listed in Section 9.2 of the EP (USEC, 2006a). The recovery organization will be managed by a Recovery Manager, who has overall responsibility for recovery activities including assessing damage to the facility and ensuring that safety equipment will be checked and restored to normal conditions.

The staff reviewed the applicant's safe shutdown process against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.12, and, based on its review, the staff finds that the EP (USEC, 2006a) describes:

1. The methods and responsibilities for assessing the damage to, and status of the facility;

2. The procedures for determining the actions necessary to reduce any ongoing releases of materials and to prevent further incidents;

3. The provisions for promptly beginning restoration actions; and

4. Key positions in the recovery organization.

8.3.13 Exercises and Drills

Section 7.3 of the EP (USEC, 2006a) describes the provisions in place to conduct quarterly communications checks and biennial exercises to develop, maintain, and test the response capabilities of emergency personnel, facilities, equipment, procedures, and training. The Plant Support Manager has overall responsibility for implementing a coordinated program of emergency drills and exercises identified in a procedure.

An exercise scenario manual will be prepared for each drill and exercise. These scenarios will be varied annually. Drills develop, test, or maintain a specific emergency response capability using a limited scope scenario, whereas exercises test the integrated capability of all or most of the basic elements in the EP (USEC, 2006a) and procedures. The applicant has committed that no scenario information will be provided to participants before an exercise. Off-site response organizations will be invited to observe or participate in the biennial exercises.

Drill and exercise controllers and evaluators will be trained on safety precautions, scenario messages, simulated actions, participant interactions and controller input, evaluation methodology, and critique format.

As discussed in Section 7.4 of the EP (USEC, 2006a), formal critiques will be conducted after each drill and exercise. Critiqued items that have safety significance, indicate a regulatory violation, or reflect serious deficiencies in plan content or implementation will be identified and documents using the Corrective Action Program.

The applicant will conduct quarterly communications checks with off-site response organizations. This includes checking and updating telephone numbers, as necessary.

The staff reviewed the applicant's exercise and drills against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.13, and, based on its review, the staff finds that the EP (USEC, 2006a) contains commitments to conduct exercises and drills in a manner that demonstrates the capability of the organization to plan and perform an effective response to an emergency. The EP (USEC, 2006a) demonstrates the following:

1. Task-related knowledge will be demonstrated through periodic participation by all qualified individuals for each position in the ERO;

2. Drill performance will be assessed against specific scenario objectives, using postulated accidents, that adequately test personnel, equipment, and resources;

3. Effective player, controller, evaluator, and observer pre-drill briefings will be conducted;

4. Scenario data and exercise messages will provided by the controllers in a manner that will not interfere with the drill performance;

5. Trained evaluators will be used to identify and record participant performance, scenario, strengths and deficiencies, and equipment problems;

6. Pre-staging of equipment and personnel will be minimized to realistically test the activation and staffing of emergency facilities;

7. Critiques will be conducted in a timely manner;

8. Emergency drills will demonstrate that resources are effectively used to control the site, mitigate further damage, control radiological releases, perform required on-site activities under simulated radiation/airborne and ether emergency conditions, accurately assess the facility's status during an accident, and initiate recovery;

9. Emergency drills will demonstrate personnel protection measures, including controlling and minimizing hazards to individuals during fires, medical emergencies, mitigation activities, search and rescue, and other similar events;

10. The emergency drills will demonstrate that on-site communications effectively support emergency response activities;

11. Emergency drills will demonstrate that the emergency public information organization disseminates accurate, reliable, timely, and understandable information;

12. Provisions have been made for conducting quarterly communications checks with off-site response organizations; and

13. Off-site organizations will be invited to participate in the biennial on-site exercise.

8.3.14 Responsibilities for Developing and Maintaining the Emergency Program and Its Procedures Current

Section 7 of the EP (USEC, 2006a) describes the responsibilities for developing, maintaining, and updating the Emergency Program. The applicant will maintain and update the Emergency Program and distribute it to groups having responsibilities for response functions in accordance with 10 CFR 30.34(f), 10 CFR 40.35(f), and 10 CFR 70.32(i). The applicant may change the Emergency Program without receiving prior NRC approval, provided the change does not decrease the effectiveness of the Emergency Program. The applicant will provide NRC and affected off-site response organizations with copies of any changes within six months of the change.

The staff reviewed the applicant's responsibilities for developing and maintaining the emergency program and its procedures current against the acceptance criteria in NUREG-1520 (NRC, 2002), Section 8.4.3.1.14, and, based on its review, the staff finds that the applicants EP (USEC, 2006a) describes:

1. The means for ensuring that revisions to the Emergency Program and the procedures used to implement the Emergency Program will be adequately prepared, kept up to date, and distributed to all affected parties, including NRC; and

2. The provision for approving the implementing emergency procedures, making and distributing changes to the procedures, and ensuring that each person responsible for an emergency response function has immediate access to a current copy of emergency procedures.

8.4 EVALUATION FINDINGS

The staff has evaluated the EP (USEC, 2006a) for the facility. In accordance with 10 CFR 30.32(i), 10 CFR 40.31(j), and 10 CFR 70.22(i), the applicant has established an EP (USEC, 2006a) for responding to the radiological hazards resulting from a release of radioactive material or hazardous chemicals incident to the processing of licensed material. NRC staff reviewed the EP (USEC, 2006a) with respect to 10 CFR 30.32(i), 10 CFR 40.31(j), and 10 CFR 70.22(i) and the acceptance criteria in Section 8.4.3 of NUREG-1520 (NRC, 2002). The staff determined that the applicant's EP (USEC, 2006a) is adequate to demonstrate compliance with 10 CFR 30.32(i), 10 CFR 40.31(j), and 10 CFR 70.22(i), in that: (1) the facility is properly configured to limit releases of radioactive materials in case of an accident; (2) a capability exists for measuring and assessing the significance of accidental releases of radioactive materials; (3) appropriate emergency equipment and procedures will be provided on-site, to protect workers against radiation and other chemical hazards that might be encountered after an accident; (4) a system has been established to notify Federal, State, and local Government agencies and to recommend appropriate protective actions to protect members of the public; and (5) necessary recovery actions will be established to return the facility to a safe condition after an accident.

8.5 REFERENCES

(ANSI/ANS, 1997) American National Standards Institute/American Nuclear Society (ANSI/ANS). ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response," 1997.

(EPA, 1992) U.S. Environmental Protection Agency (EPA). "Manual of Protective Action Guidelines and Protective Actions for Nuclear Incidents," 1992.

(NRC, 1992) U.S. Nuclear Regulatory Commission (NRC). Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," 1992.

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

(USEC, 2005) USEC Inc. (USEC). "Portsmouth Gaseous Diffusion Plant Emergency Plan," Revision 79, August, 2005.

(USEC, 2006a) USEC Inc. (USEC). "Emergency Plan for the American Centrifuge Plant in Piketon Ohio," Revision 10, August 2006.

(USEC, 2006b) USEC Inc. (USEC). "License Application for the American Centrifuge Plant in

Piketon, Ohio," Revision 18, September 2006.

9.0 ENVIRONMENTAL PROTECTION

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's environmental protection plan is to evaluate whether the application provides environmental protection measures that are adequate to protect the environment, and the health and safety of the public as required by 10 CFR Parts 20, 30, 40, 51, and 70.

9.1 REGULATORY REQUIREMENTS

For its plan to be considered acceptable, the applicant must satisfy the following regulatory requirements regarding environmental protection:

1. 10 CFR Part 20 specifies the effluent control and treatment measures necessary to meet the dose limits and dose constraints for members of the public specified in: (a) Subparts B, D, and F; (b) the survey requirements of Subpart F; (c) the waste disposal requirements of Subpart K; (d) the records requirements of Subpart L; and (e) the reporting requirements of Subpart M.

2. 10 CFR 30.33 specifies in part that an application for the possession and use of byproduct material will be granted provided that, among other things, the applicant's proposed equipment and facilities are adequate to protect health and minimize danger to life or property, and that the applicant is qualified by training and experience to use the byproduct material for the purpose requested in such a manner as to protect health and minimize danger to life and property.

3. 10 CFR 40.31(k) states that a license application (LA) for a uranium enrichment facility must be accompanied by an Environmental Report (ER) required under Subpart A of Part 51.

4. 10 CFR 40.32(e) states that,

   In the case of an application for a license for a uranium enrichment facility, or for a license to possess and use source and byproduct material for uranium milling, production of uranium hexafluoride, or for the conduct of any other activity which the Commission determines will significantly affect the quality of the environment, the Director of Nuclear Material Safety and Safeguards or his designee, before commencement of construction of the plant or facility in which the activity will be conducted, on the bass of information filed and evaluations made pursuant to Subpart A of Part 51 of this chapter, has concluded, after weighing the environmental, economic, technical and other benefits against environmental costs and considering available alternatives, that the action called for is the issuance of the proposed license, with any appropriate conditions to protect environmental values. Commencement of construction before this conclusion is grounds for denial of a license to possess and use source and byproduct material in the plant or facility. As used in this paragraph, the term 'commencement of construction' means any clearing of land, excavation, or other substantial action that would adversely affect the environment of a site. The term does not mean site exploration, roads necessary for site exploration, borings to determine

foundation conditions, or other preconstruction monitoring or testing to establish background information related to the suitability of the site or the protection of environmental values.

5. 10 CFR Part 51 specifies that the applicant must submit an ER for construction and operation of a uranium enrichment facility, as required by 10 CFR 51.60(b)(1)(vii).

6. 10 CFR 70.22(a)(7) specifies that the applicant must provide a description of the equipment and facilities that the applicant will use to protect health and minimize danger to life and property.

7. 10 CFR 70.59 outlines the radiological effluent monitoring reporting requirements for a 10 CFR Part 70 licensee.

8. 10 CFR 70.65(b) specifies, among other things, that an applicant for a facility must provide an Integrated Safety Analysis (ISA) Summary that includes a list of the Items relied on for safety (IROFS) established by the applicant to reduce the risks of credible high-consequence events and credible intermediate-consequence events.

9.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the environmental protection program description section of the LA (USEC, 2006c) is contained in Chapter 9 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). The acceptance criteria applicable to this review are contained in Section 9.4.3.2. Other acceptance criteria in Chapter 9 of NUREG-1520 (NRC, 2002) relating to environmental documents developed pursuant to the National Environmental Policy Act (NEPA) are not applicable. These NEPA-related acceptance criteria apply to evaluating the adequacy of the applicant's ER and to determining whether the staff should prepare an environmental assessment (EA) or an environmental impact statement (EIS) pursuant to regulatory requirements found in 10 CFR Part 51. The staff prepared a final EIS in April 2006, "Final Environmental Impact Statement for the Proposed American Centrifuge Plant in Piketon, Ohio," NUREG-1834 (NRC, 2006).

9.3 STAFF REVIEW AND ANALYSIS

9.3.1 Radiation Safety

9.3.1.1 As Low As is Reasonably Achievable (ALARA) Goals for Air and Liquid Effluent Control

The applicant has stated that it will maintain and use gaseous and liquid treatment systems, as appropriate, to maintain releases of radioactive material to unrestricted areas below the limits specified in 10 CFR 20.1301 and 10 CFR Part 40, and in accordance with ALARA policy. Unrestricted areas are those areas beyond U.S. Department of Energy (DOE) reservation boundary and to which any member of the public has unrestricted access. The applicant's approach is sufficiently detailed to demonstrate that the applicant is in compliance with regulatory dose limits found in 10 CFR 20.1301, that the air and liquid dose constraints are within the guidance found in Section 9.4.3.2.1(1) of NUREG-1520 (NRC, 2002), and that the applicant's ALARA program for controlling gaseous and liquid effluents is within the guidance

found in NRC Regulatory Guide 8.37 (NRC, 1993).

Air Effluent ALARA Goal

For radiological ALARA goals for air effluent control, the applicant proposes an ALARA goal for the American Centrifuge Plant (ACP) of 5 percent (0.5 mrem/year) of the 10 CFR 20.1101 constraint of 10 mrem/year for the maximally exposed member of the public. The applicant's proposal is less than the 10 mrem/year ALARA goal recommended in NRC Regulatory Guide 8.37 (NRC, 1993), Regulatory Position C.1.2, "ALARA Goals," and is, therefore, acceptable to the staff. The applicant's approach also is in general agreement with the acceptance criterion found at 9.4.3.2.1(1) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable to the staff.

Liquid Effluent ALARA Goal

For liquid effluents, the applicant proposes an ALARA goal of 10 percent of the air effluent goal, or 0.05 mrem/year to the maximally exposed member of the public. This is equivalent to 0.05% of the 10 CFR 20.1301 limit on annual public dose.

The applicant's proposal is much less than the 10 mrem/year goal recommended in NRC Regulatory Guide 8.37 (NRC, 1993), Regulatory Position 1.2 (ALARA Goals), is in general agreement with the guidance found in Section 9.4.3.2.1(2) of NUREG-1520 (NRC, 2002), and is, therefore, acceptable to the staff.

9.3.1.2 Air Effluent Controls to Maintain Public Doses ALARA

Section 4.6.3.2 of the ER (USEC, 2006a) generally describes several effluent controls to reduce emissions of radioactivity to the atmosphere to maintain doses to the public ALARA, following the guidance found in Section 9.4.3.2.1(2) of NUREG-1520 (NRC, 2002). These controls include:

- Cold traps desublime uranium hexafluoride and separate it from other gases followed by activated alumina traps to capture uranium hexafluoride in the purge vacuum (PV) and Evacuation Vacuum (EV) Systems;

- A continuous vent sampler that draws samples from the process vent using an isokinetic probe that maintains a real time indication of effluent levels; and

- Engineered local ventilation systems to capture residual uranium during maintenance activity around the centrifuges.

In Section 1.1 of the LA (USEC, 2006c), the applicant describes the facilities and buildings that comprise the ACP, and in Section 9.2.1.2.1 of the LA (USEC, 2006c), the applicant identifies specific controls for airborne effluents. Those buildings and associated controls are described below.

X-3346 Feed and Customer Services Building

The Feed Area of X-3346 contains a variety of potential sources for radioactive effluents, both as gaseous uranium hexafluoride and particulate uranyl fluoride. These sources are vented to the atmosphere through an evacuation system, which has separate subsystems to control gaseous and airborne particulate effluents. The subsystems exhaust to a continuously monitored combined vent. Quality control sampling and transfer of uranium hexafluoride to customer cylinders for shipment are conducted in the Customer Services Area of X-3346, which is also a source of the same radioactive effluents, which are vented through a similar evacuation system.

Gases are processed through cold traps to capture the uranium hexafluoride; any residual is further reduced by passing the gas through an alumina trap. Uranium hexafluoride released from cylinder connections and disconnections that reacts with humid air to create uranyl fluoride is captured by gulper systems and then passed through a roughing filter followed by a High Efficiency Particulate Air filter to collect the uranyl fluoride particulate.

X-3001 and X-3002 Process Buildings

The process buildings contain the centrifuge machines that separate the uranium hexafluoride into enriched product and depleted tails, and contain potential sources for gaseous radioactive effluents. These sources are vented to the atmosphere through either the PV or EV Systems, which in turn exhaust to a common, continuously monitored vent. The PV/EV Systems remove air in the enrichment equipment, which may have traces of uranium hexafluoride gas, and pass it through shared alumina traps before venting.

X-3356 Product and Tails Withdrawal Building

The X-3356 building withdraws and desublimes both the product and tail streams from the enrichment process and contains a variety of sources for radioactive effluents, both as gaseous uranium hexafluoride and particulate uranyl fluoride. These sources are vented to the atmosphere through evacuation systems similar to those for the X-3346 building described above.

X-3012 Process Support Building

The X-3012 building provides process control functions and maintenance support. The applicant indicated that from time to time, contaminated components may be serviced in the maintenance shops in this building. Engineered local ventilation systems may be used to collect any residual uranium.

X-7725 Recycle/Assembly Facility, X-7726 Centrifuge Training and Test Facility, and X-7727H Interplant Transfer Corridor

Centrifuges are to be assembled and may be disassembled for repair or inspection in either the X-7725 or X-7726 facilities. Centrifuges requiring repair or examination may use engineered local ventilation systems to capture any residual uranium. The applicant indicated that assembled centrifuge machines will be tested with uranium hexafluoride in Gas Test Stands located in a separate room within the X-7725 facility containing its own ventilation and emission control system. Exhaust from the test stands will pass through alumina traps to a continuously monitored vent.

The X-7727H corridor is used only to provide indoor transport for individual centrifuges between the X-7725 facility and the process buildings and is not exposed to a source of gaseous uranium.

Staff Evaluation of Air Effluent Controls

The staff evaluated the air effluent controls described in Section 9.2.1.2.1 of the LA (USEC, 2006c) and radiological air quality described in Section 4.6.3.2 of the ER (USEC, 2006a). These controls are expected to result in maximum air effluent releases of about 2.7 mCi per week, or about 0.14 Ci per year of total uranium, resulting in a projected maximum airborne concentration of uranium of less than 3.2×10^{-15} µCi/mL, with an associated total effective dose equivalent (TEDE) of about 0.3 mrem to the Maximally Exposed Individual (MEI). Applicant calculations using the CAP88-PC model indicate that dose rate to the MEI are will below the U.S. Environmental Protection Agency (EPA) National Emissions Standards for Hazardous Air Pollutants limit of 10 mrem/year and the NRC limit of 100 mrem/year. The staff finds that the applicant's controls will ensure that radiation levels to the public will remain well below regulatory limits and ALARA air effluent goals, that the applicant's approach to effluent controls is generally consistent with the guidance found in Section 9.4.3.2.1(2) of NUREG-1520 (NRC, 2002), and are, therefore, acceptable to the staff. Therefore, the staff finds that the applicant has demonstrated that it will reduce gaseous effluents to provide adequate protection of the environment and of the health and safety of the public.

9.3.1.3 Liquid Effluent Controls to Maintain Public Doses ALARA

Section 9.2.1.2.2 of the LA (USEC, 2006c) describes several liquid effluent controls to maintain doses to the public ALARA, following the guidance found in Section 9.4.3.2.1(2) of NUREG-1520 (NRC, 2002).

In Section 9.2.1.2.2 of the LA (USEC, 2006c), the applicant indicates that the centrifuges and PV/EV vacuum pumps are to be cooled by a closed-loop Machine Cooling Water (MCW) system to minimize the amount of water potentially contaminated by uranium. Waste heat from the MCW system will be discharged via heat exchangers to the Tower Water Cooling (TWC) system. Waste heat from the cold trap refrigeration systems in the X-3346 and X-3356 buildings will also be discharged to the TWC system. The applicant proposes to use the Gaseous Diffusion Plant (GDP) Recirculating Cooling Water (RCW) system to discharge blow down water from the TWC system, as is currently done. No licensed material is expected in this location. The applicant indicated that at some time in the future, the GDP will be decommissioned. Before the GDP is decommissioned, the applicant will bypass the GDP RCW system. Instead, the effluent will be discharged directly into the RCW discharge pipeline and then into the Scioto River. This change will not have any effect because no treatment of the effluent occurs in the RCW process.

The applicant states in Section 9.2.2.1.2 of the LA (USEC, 2006c) that the liquid discharges to the RCW System are monitored by using an automated sampler, which collects a weekly composite sample of the liquid effluent for radiological analysis as well as samples for NPDES-mandated analyses. These data are made available to the ACP as assurance that no unanticipated discharge of licensed material has occurred.

The applicant proposes to collect leaks from the MCW system and incidental spills elsewhere in

the ACP in the Liquid Effluent Collection (LEC) system. Water accumulated in the 550 gallon tank is sampled and pumped to either the X-6619 sewage treatment plant (STP) or containerized for disposal, depending on the results. Given the small increment in waste volumes from the MCW and existing compliance at the X-6619 STP outfall, this proposal to discharge the MCW system to this outfall is acceptable to the staff.

The applicant states in Section 9.2.2.2.1 of the LA (USEC, 2006c) that no changes in storm water runoff associated with installation and operation of the ACP are expected. Storm water runoff in the vicinity of the ACP is captured in either the X-2230N West Holding Pond (National Pollutant Discharge Elimination System (NPDES) outfall 012) or the X-2230M Southwest Holding Pond (NPDES outfall 013). Table 9.2-4, "Anticipated Liquid Effluents," (USEC, 2006c) states that each holding pond contains maximum anticipated liquid discharge concentrations of $1x10^{-8}$ µCi/mL uranium and discharges to the Scioto River.

Staff Evaluation of Liquid Effluent Controls

As stated above, the applicant does not anticipate any liquid discharges of licensed radioactive materials from the proposed facility. Based on historical operating experience at the Portsmouth reservation, the applicant has established maximum liquid effluent concentrations expected under normal operating conditions. Table 9.2-4 of the LA (USEC, 2006c) lists these anticipated concentrations along with corresponding release limits from 10 CFR Part 20, Appendix B, for comparison. In all instances, the anticipated radionuclide releases are below the NRC regulatory limits. Any discharges to the Scioto River would be below regulatory limits before any dilution provided by the river.

The staff finds that the applicant's effluent controls will ensure that radiation levels will remain well below regulatory limits and ALARA liquid effluent goals, and that the applicant's approach to liquid effluent controls is generally consistent with the guidance found in Section 9.4.3.2.1(2) of NUREG-1520 (NRC, 2002), thus, the applicant's controls and approach to the controls are acceptable to the staff. Therefore, the staff finds that the applicant has demonstrated that it will reduce liquid effluents to provide adequate protection of the environment and of the health and safety of the public.

9.3.1.4 ALARA Reviews and Reports to Management

Section 9.2 of the LA (USEC, 2006c) describes the applicant's ALARA reviews and reports to management. To maintain effluents below regulatory limits and ALARA goals, the applicant proposes to implement a program based on the use of a Baseline Effluent Quantity (BEQ) for uranium and technetium. The BEQ is defined as the maximum effluent expected under normal operating conditions. For the ACP, the new BEQ values are 0.2 mCi per week uranium and 0.1 mCi per week technetium for the ACP Process Vents, and $2.5x10^{-8}$ µCi/mL uranium and $1.0x10^{-7}$ µCi/mL technetium for the X-2230N West Holding Pond, and for the X-2230M Southwest Holding Pond.

The applicant has defined additional action levels above the BEQ. For uranium, effluent levels between 2 and 10 times the BEQ will trigger an investigation into the causes of increased releases by either an individual weekly uranium release equal to or greater than 10 times the BEQ or a rolling average of the weekly uranium releases over the previous 6 months equal to or greater than 2 times the BEQ. For releases above an EPA Reportable Quantity (>0.1 Ci in 24

hours), notification of the Plant Shift Superintendent and control of the source or shutdown is specified. Above 1 Ci per week, corresponding to 2 mrem at the maximally exposed individual off site, the applicant will close the affected vents until control of emissions is re-established. These levels are incremental, resulting in increasingly more aggressive action before a regulatory limit is exceeded, and are therefore acceptable to the staff.

The required action the applicant describes for releases above a BEQ include the ACP Shift Supervisor and Plant Shift Superintendent notifying management and initiating an evaluation of whether additional emission controls would significantly reduce public exposure. BEQs are reviewed annually, at a minimum, to ensure the principles described in the applicant's ALARA policy are followed. This review also includes analyses of trends in radioactive effluents and environmental monitoring data. Table 9.2-1 of the applicant's LA (USEC, 2006c) identifies action levels for radionuclide effluents that may result in additional controls (mitigative or corrective operational changes) being applied to reduce public exposure. Results of the review are reported to the ACP Regulatory Manager and other senior management as described in Chapter 4 of the applicant's LA (USEC, 2006c). This approach is consistent with the guidance in Section 9.4.3.2.1(3) of NUREG-1520 (NRC, 2002), and is therefore acceptable to the staff.

9.3.1.5 Waste Minimization

Section 4 of the applicant's ER (USEC, 2006a) describes facility features and systems that will minimize the generation of radioactive waste. Section 9.2.1.4 of the applicant's LA (USEC, 2006c) describes the applicant's waste minimization activities. Individual waste streams are identified and characterized based on: (a) process knowledge; (b) routine radiation surveys as described in Chapter 4 of the LA (USEC, 2006c); and (c) laboratory analysis, as needed. Waste management costs are tracked through a formal Request-for-Disposal database system administered by waste management personnel and the annual budgeting process.

Radioactive waste minimization activities have the support of the applicant's senior management. The applicant provides environmental and waste management professionals with opportunities to attend off-site training and conferences to exchange technical information on waste minimization. The applicant indicated that it will assign the responsibility for coordinating waste minimization activities at the ACP to the ACP Regulatory Manager.

Proposed waste minimization practices would include, for example, the promotion of the use of non-hazardous materials, process optimization, in-process closed-loop recycle, and waste segregation. The applicant emphasizes the minimization of hazardous waste, mixed waste, and low-level radioactive waste generation. These systems are consistent with the guidance provided in NRC Information Notice 94-23, "Guidance to Hazardous, Radioactive, and Mixed-Waste Generators on the Elements of a Waste Minimization Program" (NRC, 1994). As described more fully in Chapter 4 of this SER, the ALARA program is acceptable to the staff.

The applicant's program complies with waste minimization requirements found in 10 CFR 20.1406, is consistent with the guidance in Section 9.4.3.2.1(4) of NUREG-1520 (NRC, 2002), and is therefore acceptable to the staff.

9.3.1.6 Safe Handling of Radioactive Waste

The applicant has identified a system of waste packaging, labeling, storage, treatment,

shipments and disposal for solid and liquid waste. Waste would be packaged to meet U.S. Department of Transportation (DOT) and 10 CFR Part 71 requirements.

With respect to storage, the applicant states that the State of Ohio has provided an exemption to the 90-day limit for any ACP mixed waste. Where outdoor storage is required, waste with removable contamination would be packaged in containers and wrapped or covered to prevent the release of radioactivity. Mixed aqueous waste that cannot be processed in the applicant's facilities would be stored on site until treatment is available at commercial treatment facilities that are licensed in accordance with 10 CFR Part 61, or NRC Agreement State requirements.

Liquid low-level radioactive waste and mixed waste solutions requiring treatment are containerized for disposal. Processed wastewater would be released to the X-6619 STP.

Off-site shipments of radioactive wastes are manifested in accordance with 10 CFR 20.2006 requirements and are packaged, labeled, and manifested in accordance with applicable State, DOT, NRC and EPA requirements. Disposal is in compliance with 10 CFR Part 20, Subpart K requirements and records are retained in accordance with 10 CFR 20.2108. Classified waste would be disposed of in accordance with 10 CFR Part 95 and Security program requirements.

9.3.2 Effluent and Environmental Monitoring

The applicant describes its environmental monitoring program for radiological and non-radiological releases from the proposed ACP in Chapter 6 of its ER (USEC, 2006a) and in greater detail in Section 9.2.2 of its LA (USEC, 2006c). A detailed NRC staff evaluation of the applicant's program is found in Chapter 6 of the Final EIS (NRC, 2006).

9.3.2.1 Air Effluent Monitoring

The applicant describes its compliance program for Subpart H of 40 CFR Part 61, "National Emission Standards for Hazardous Air Pollutants," in Section 9.2.2.1.2 of the LA (USEC, 2006c). These standards are consistent with 10 CFR 20.1101(d), which forms the basis for the applicant's air effluent ALARA program goals. As described in SER Section 9.3.1.2, the expected concentrations of radioactive materials in air effluents are ALARA and are below the limits specified in 10 CFR Part 20, Appendix B, Table 2.

With regard to compliance with NRC standards for public dose in 10 CFR 20.1301, the applicant will calculate TEDE to the maximally exposed person. The applicant states in Section 9.2.2.2.1 of the LA (USEC, 2006c) that it will use the CAP88 package of computer codes, which implements the downwind dispersion model from Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluent," (NRC, 1997) for human intake of airborne radioactivity, ground deposition of airborne radioactivity, uptake of radioactivity in foodstuff, and consumption of contaminated food. This approach is acceptable to the staff.

The applicant has identified the locations of two potential sources of radioactivity discharges to air from the ACP process: (1) the EV and PV systems discharged through the X-3001 process vent; and (2) fugitive emissions from the X-7726 static stand captured by local ventilation systems. The applicant does not expect measurable emissions from the X-7726 static stand as a result of opening centrifuges that have operated on uranium hexafluoride gas.

A continuous vent sampler using alumina media would be used to monitor the EV and PV system vents for uranium hexafluoride. Weekly primary sample traps would be analyzed for uranium-234, uranium-235, uranium-238, and technetium-99 (Tc-99). A secondary trap would be replaced quarterly. The applicant does not expect to detect technetium-99 in the ACP, but all vent samplers at Portsmouth Gaseous Diffusion Plant (PORTS), including those at the ACP, will be analyzed for Tc-99. Uranium isotope concentrations would be determined using either alpha spectroscopy or Inductively Coupled Plasma/Mass Spectrometry, with minimum detectable activity less than 0.2 percent of the 10 CFR Part 20, Appendix B, Table 2 values. A representative sample of air effluent is collected using an isokinetic probe and monitoring of both vent and sampler air flows.

A number of buildings have similar air monitoring and effluent controls (e.g., alumina traps, continuous vent sampler, ventilation systems to capture residual uranium). These buildings include: the Feed and Customer Services Building (X-3346); the Process Buildings that house the centrifuges (X-3001 and X-3002); the Product and Tails and Withdrawal Building (X-3356); and, to a lesser extent, the Process Support Building (X-3012).

On the basis of this analysis, the staff finds that the applicant's air effluent monitoring program ensures that concentrations of radioactivity in air effluent will be below limits in 10 CFR Part 20, is consistent with the guidance in Section 9.4.3.2.2(1) of NUREG-1520 (NRC, 2002), is consistent with guidance in Regulatory Guide 1.109 ((NRC, 1997), and is, therefore, acceptable to the staff.

9.3.2.2 Liquid Effluent Monitoring

The applicant's liquid effluent monitoring program is described in Section 9.2.2.2 of the LA (USEC, 2006c). As described in SER Section 9.3.1.2, 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. With regard to compliance with NRC standards for public dose in 10 CFR 20.1301, the applicant will calculate TEDE to the maximally exposed person downstream of plant discharges on the Scioto River. To calculate TEDE, the applicant will use the Regulatory Guide 1.109 (NRC, 1997) model, with consideration of direct use of contaminated surface water for drinking water, the ingestion of fish from contaminated water, and the use of contaminated water for irrigation. This approach addresses the major pathways for uranium, and is therefore acceptable to the staff.

The applicant identified the locations of four potential sources of radioactivity discharges to surface water from the ACP process, including four NPDES outfalls: (1) X-6619 Sewage Treatment Plant identified as NPDES outfall 003; (2) GDP RCW System identified as NPDES outfall 004; (3) X-2230N West Holding Pond identified as NPDES outfall 012; and (4) X-2230M Southwest Holding Pond identified as NPDES outfall 013.

The applicant will use composite samplers and weekly analysis to measure contaminants in leased outfalls that have continuous flow. Outfalls with intermittent flow would be monitored with grab samplers during periods of outfall flow. The applicant will monitor uranium, gross alpha, gross beta, and technetium beta radioactivity. Uranium radioactivity would be measured by Inductively Coupled Plasma/Mass Spectrometry, with minimum detectable concentrations of 0.001 µg per milliliter, which results in a minimal detection concentration (MDC) below 2 percent of the applicable 10 CFR Part 20, Appendix B, Table 2 values. An MDC of less than 5 percent

is acceptable to the staff.

With respect to permits from other agencies, the applicant is required under Condition II.K of the NPDES Permit to submit a quarterly written report to the State of Ohio EPA summarizing the radioactive discharges from the permitted outfalls. A copy of this report is to be submitted to NRC within 60 days of the end of the calendar quarter.

The applicant will use level gauges to detect unplanned releases to ground water or soil from the Liquid Effluent Control underground tanks.

On the basis of this analysis, the staff finds that the applicant's liquid effluent monitoring program ensures that concentrations of radioactivity in liquid effluent will be below limits in 10 CFR Part 20, is consistent with the guidance in Section 9.4.3.2.2(1) of NUREG-1520 (NRC, 2002), is consistent with the guidance in Regulatory Guide 1.109 (NRC, 1997), and is, therefore, acceptable to the staff.

9.3.2.3 Laboratory Quality Control

The staff has reviewed Section 9.2.2.5 of the applicant's LA (USEC, 2006c) regarding laboratory standards. The staff also reviewed the ACP Quality Assurance Program Description (USEC, 2006d) and found it acceptable (see Chapter 11 of this SER). Environmental thermoluminescence dosimeters are processed by a vendor certified by the National Voluntary Laboratory Accreditation Program. A laboratory licensed by NRC or an Agreement State provides other radiological and chemical analyses for the monitoring and measurement program. The on-site X-710 Building Analytical Laboratory is licensed by the State of Ohio and certified by NRC, but is not part of the proposed ACP or operated by the applicant. At the X-710 building Analytical Laboratory, the applicant will use dedicated Chain of Custody procedures, National Institute of Standards and Technology traceable standards, matrix spikes, duplicate and replicate samples, check samples, and blind and double blind quality control samples as quality control practices. In addition, the applicant will continue participation in external control programs. The staff finds that these procedures are consistent with the guidance in Section 9.4.3.2.2(1)(h) of NUREG-1520 (NRC, 2002), and are adequate to validate the analytical results of the Analytical Laboratory.

9.3.2.4 Environmental Monitoring

The applicant's environmental monitoring program is described in detail in a number of other documents. The applicant has described the existing PORTS environmental monitoring program in its ER (USEC, 2006a) in the context of the low levels of radioactive effluents expected from the proposed ACP. The environmental monitoring program is also described in section 9.2.2 of the LA (USEC, 2006c). The staff's review of the applicant's program can also be found in chapter 6 of the Final EIS, which was published in April 2006 (NRC, 2006). On the basis of these low expected levels of radioactive effluents, the applicant has not proposed significant changes to the PORTS environmental monitoring program. The applicant will demonstrate compliance with NRC radiation standards at the proposed ACP by using effluent monitoring and atmospheric dispersion modeling. DOE will continue to be responsible for legacy groundwater contamination that resulted from operation of its facilities, including the PORTS trichloroethylene plumes. Given the low levels of effluent expected from the ACP, and the small increment the ACP would contribute to existing PORTS operations, the staff finds the

applicant's environmental monitoring program and its proposal to use effluent monitoring and modeling to demonstrate compliance with the regulations of 10 CFR Parts 20 and 70 to be acceptable and consistent with the guidance in Section 9.4.3.2.2(2) of NUREG-1520 (NRC, 2002).

9.3.3 ISA Summary

The staff provides its evaluation of the ISA Summary (USEC, 2006b) in Chapter 3 and Appendix A of this SER. Chapter 3 evaluates the ISA methodology and finds the methodology to be satisfactory. Appendix A states that the ISA Summary (USEC, 2006b) is complete, provides reasonable estimates of the likelihood and consequences of each accident sequence, and provides sufficient information to determine whether adequate engineering or administrative controls are identified for each accident sequence. Chapter 11 of this SER contains the staff's evaluation of management measures used to ensure that IROFS will satisfactorily perform their intended safety functions. The staff verified that environmental release limits would be met using existing IROFS. Therefore, no additional IROFS are identified for the ACP for reducing the environmental risks of natural phenomena and potential accidents.

Under 10 CFR Part 70, Subpart H (sections 70.60 through 70.76), the applicant is to assure, among other things, compliance with various performance requirements to reduce the risks of credible high-consequence events (i.e., accidents) and credible intermediate-consequence events. 10 CFR 70.6(c)(3) identifies the environmental performance requirement that the applicant apply controls such that a credible intermediate-consequence event is unlikely to occur or that the consequence of such an event will not exceed a 24 hour averaged release of radioactive material outside the restricted area in concentrations 5000 times the values in 10 CFR Part 20, Appendix B, Table 2.

To reduce the risks of accidents, the applicant evaluated accident sequences and applied appropriate preventive and mitigative measures, as needed. The applicant identified various sequences in the ISA Summary (USEC, 2006b) for radiological and non-radiological accidents, which were evaluated to assure adequate protection of worker health and safety. By assuring that all credible high-consequence events are rendered highly unlikely and that all intermediate-consequence events are rendered unlikely (i.e., event sequences are prevented), the applicant also assured that the environmental performance requirements of 10 CFR 70.61(c)(3) will be met. However, for accident sequences where the applicant applied IROFS to mitigate the consequences to below the radiological and environmental performance requirements, the applicant calculated the environmental consequences based on a 24 hour concentration of uranium at the restricted area boundary.

As noted above, the applicant's approach to risk reduction is to be accomplished through a combination of preventive and mitigative measures, with the emphasis on preventive measures. A more complete discussion is found in Chapter 3 and Appendix A of this SER, which address credible high- and intermediate-consequence accident sequences. It also addresses preventive and mitigative measures.

The staff finds that the applicant's ISA Summary (USEC, 2006b) complies with 10 CFR Part 70, is consistent with the guidance in Section 9.4.3.2.3 of NUREG-1520 (NRC, 2002), and is therefore acceptable to the staff.

9.4 EVALUATION FINDINGS

The applicant has developed a program to implement adequate environmental protection measures, including: (1) environmental and effluent monitoring; and (2) effluent controls to maintain public doses ALARA as part of its radiation protection program. NRC staff concludes that the applicant's program, as described in its application, is adequate to protect the environment, and the health and safety of the public. The applicant's environmental monitoring program for the ACP complies with the regulatory requirements imposed by the Commission in 10 CFR Parts 20, 30, 40, 51, and 70, is consistent with the guidance in NUREG-1520 (NRC, 2002), and is therefore acceptable to the staff.

NRC staff issued a Final EIS in April 2006 (NRC,2006) for this licensing action, as required by 10 CFR 51.20. After weighing the environmental impacts of the proposed construction, operation, and decommissioning of the proposed facility and comparing alternatives, NRC staff recommends in its Final EIS that, unless safety issues mandate otherwise, the proposed license be issued to USEC.

9.5 REFERENCES

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

(NRC, 1994) U.S. Nuclear Regulatory Commission (NRC). Information Notice 94-23, "Guidance to Hazardous, Radioactive, and Mixed-Waste Generators on the Elements of a Waste Minimization Program," 1994.

(NRC, 1997) U.S. Nuclear Regulatory Commission (NRC). Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix 1," 1997.

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

(NRC, 2006) U.S. Nuclear Regulatory Commission (NRC). NUREG-1834, "Final Environmental Impact Statement for the Proposed American Centrifuge Plant in Piketon, Ohio," 2006.

(USEC, 2006a) USEC Inc. (USEC). "Environmental Report for the American Centrifuge Plant in Piketon, Ohio," Revision 8, April, 2006.

(USEC, 2006b) USEC Inc. (USEC). "Integrated Safety Analysis Summary for the American Centrifuge Plant in Piketon, Ohio," Revision 14, August 2006.

(USEC, 2006c) USEC Inc. (USEC). "License Application for the American Centrifuge Plant in Piketon, Ohio," Revision 18, September 2006.

(USEC, 2006d) USEC Inc. (USEC). "Quality Assurance Program Description for the American Centrifuge Plant in Piketon, Ohio," Revision 3, August 2006.

10.0 DECOMMISSIONING

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's decommissioning plan is to evaluate whether the application provides for decommissioning the facility safely and in accordance with NRC requirements.

At the time of the initial license application (LA) for a uranium enrichment facility, the applicant is required to submit a decommissioning funding plan (DFP). The purpose of 's review of the 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. The DFP, therefore, should contain an overview of the applicant's proposed decommissioning activities, the methods used to determine the cost estimate, and the financial assurance mechanism. This overview should contain sufficient detail to enable the reviewer to determine whether the decommissioning cost estimate is reasonably accurate.

10.1 REGULATORY REQUIREMENTS

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

10 CFR 20.1401-1406	"Radiological Criteria for License Termination"
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 40.14	"Specific Exemptions"
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 70.17	"Specific Exemptions"
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"

10.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the decommissioning section of the LA (USEC, 2006c) is contained in Chapter 10 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002a) and in Volume 3 of "Consolidated NMSS Decommissioning Guidance," NUREG-1757 (NRC, 2003). Chapter 10 of NUREG-1520 (NRC, 2002a) is applicable to the American Centrifuge Plant (ACP), except that the review used NUREG-1757 (NRC, 2003), which is the updated version of NUREG-1727, "NMSS Decommissioning Standard Review Plan" (NRC, 2000), referenced in NUREG-1520 (NRC, 2002a). The acceptance criteria applicable to this review are contained in Section 10.4 of NUREG-1520 (NRC, 2002a).

With regard to the Acceptance Criteria, because depleted uranium deconversion services are not currently available in the United States, depleted uranium generated in the operation of the ACP facility is considered as a potential decommissioning obligation in the DFP. In addition, the applicant requested an exemption to the decommissioning funding requirements to incrementally fund its financial assurance obligation as depleted uranium is generated and centrifuge machines are built/installed. NRC staff also determined that, because the incremental funding approach proposed by the applicant will provide funding for all applicant's decommissioning obligations at any point in time, the approach will not endanger life or property or the common defense and security.

NUREG-1757 (NRC, 2003) provides guidance for developing final decommissioning plans required under 10 CFR 30.36(g), 10 CFR 40.42(g), and 10 CFR 70.38(g). A final decommissioning plan will be provided at the time of decommissioning. At the time of initial licensing and for license renewals, Section 10.1 of NUREG-1520 (NRC, 2002a) describes an overview of the proposed decommissioning activities needed to develop the DFP. This overview is a more generalized discussion of the detailed information that would be needed for the final decommissioning plan described in NUREG-1757 (NRC, 2003).

10.3 STAFF REVIEW AND ANALYSIS

NRC's staff review of the decommissioning plan focused on the applicant's conceptual decommissioning activities for the ACP, the decommissioning cost estimates, and the financial assurance for decommissioning activities. The applicant identified the decommissioning activities that may be needed in the future for decommissioning and presented site-specific estimates of decommissioning costs for those activities. Using the cost data as a basis, the applicant stated that it has presented financial assurance to cover the costs required to release the ACP for unrestricted use. The following subsections contain these decommissioning aspects as described by the applicant, and NRC staff's assessment of the applicant's proposed decommissioning plan, cost estimate, and funding plan.

Before license termination, the applicant will provide a detailed decommissioning plan that will include specific activities which will be used to protect workers, the public, and the environment.

10.3.1 Conceptual Decontamination and Decommissioning Plan

Section 10.1 of NUREG-1520 (NRC, 2002a) states that the DFP needs to contain an overview

of the proposed decommissioning activities. Section 10.3.1 of this Safety Evaluation Report (SER) provides an overview of the decommissioning activities used to develop the site-specific cost estimate of the DFP and describes the staff's review of the overview of the DFP. A detailed decommissioning plan will be provided at the time of decommissioning in accordance with 10 CFR 30.36(g), 10 CFR 40.42(g), and 10 CFR 70.38(g).

This section also addresses the recordkeeping requirements in 10 CFR 30.35(g), 10 CFR 40.36(f), and 10 CFR 70.25(g).

10.3.1.1 Decommissioning Program

There are two aspects to developing the decommissioning program. First, design features of the ACP are incorporated to control radioactive contamination. Second, features are incorporated to limit worker exposure and to control waste volume.

Section 10.1, "Decommissioning Program," of the LA (USEC, 2006c) notes that the plan for decommissioning of the ACP is to decontaminate or remove materials from the facilities promptly after cessation of operations. Special design features would be incorporated into the facility as part of decommissioning planning. Features related to radioactive contamination control include characterization of areas as clean areas, potentially contaminated areas (contamination control zones), and contaminated areas, and minimization of non-radioactive process equipment and systems in locations subject to likely contamination. These features serve to minimize the spread of radioactive contamination during operation and simplify the eventual facility decommissioning.

Section 10.1.1.2, "Worker Exposure and Waste Volume Control," of the LA (USEC, 2006c) identifies features intended to minimize worker exposure and the volume of radioactive wastes. These features include: (a) ample access for efficient equipment dismantling and removal of equipment that may be contaminated; (b) connections in the process systems for thorough purging at facility shutdown; (c) design drawings of the facility that simplify the planning and implementation of decontamination procedures; and (d) controlled worker access to contaminated areas to assure that workers wear proper protective clothing and limit their time in contaminated areas. These features not only serve to minimize worker exposure to radiation and radioactive waste volumes during decontamination activities, but also minimize the spread of any contamination.

10.3.1.2 Decommissioning Steps

Section 10.2, "Decommissioning Steps," of the LA (USEC, 2006c) describes the decommissioning methodology to be employed at the ACP, and Section 10.8 of the LA (USEC, 2006c) addresses the decontamination process. NRC's review of the applicant's decontamination process is explained in Section 10.3.1.8 of this SER.

The applicant states that implementation of decommissioning may begin immediately after facility shutdown and that the estimated time for decommissioning is 6 years. The order of steps involved for facility decommissioning would be: (a) planning and preparation (b) process system purging and system cleaning; (c) equipment dismantling and removal; (d) decontamination; (e) disposition of equipment and material (including classified items); (f) disposal of wastes; and (g) completion of a final radiation survey. The applicant indicates that a

new facility would not be required for decontamination of facility components and structures because the X-7725 facility would be the primary location for decontamination activities. The major items from the facility that are expected to require decontamination are described by the applicant in Table 10.2-1 of the LA (USEC, 2006c). These items are located inside the contamination control zones of the facility. The applicant will exercise good housekeeping practices during normal operations to maintain the other areas contamination free.

The actual decontamination method or methods to be used to decontaminate and decommission the ACP will be established before license termination based upon the site characterization survey to be performed during the decommissioning planning and preparation phase and will be described in the Decommissioning Plan.

After completing the first planning step for decommissioning, the applicant would begin process system purging and system cleaning. For this step, the applicant will remove uranium hexaflouride (UF_6) material to the extent possible by normal process operation. This would be followed by evacuation and purging of process systems. This step is estimated to take approximately 3 months.

The next step would be the equipment dismantling and removal process. The applicant describes the dismantling process as simple but labor intensive, and indicates its intention to optimize this process. For optimization, the applicant would consider several factors, described in Section 10.2.3 of the LA (USEC, 2006c). Details of the optimization process would be decided near the end of facility useful life. The applicant estimates that the time frame to accomplish dismantling and removal, as well as decontamination, is about 5 years.

Decontamination of the site requires that residual radioactivity be reduced below specified levels so that the facility may be released for unrestricted use. Decontamination of the plant will not require the installation of a new facility dedicated for that purpose. Section 10.2.4, “Decontamination,” and Section 10.8, “Decontamination,” of the LA (USEC, 2006c) describe the decontamination process. The process is addressed in Section 10.3.1.8 of this SER.

For the disposition of equipment and materials step (the sale/salvage process), the applicant states that items removed from the facility would be categorized as potentially re-usable equipment, recoverable decontaminated scrap, and wastes. Each category of items would be handled accordingly. The applicant has not assigned a salvage value to scrap material in its decommissioning funding requirement estimates. In Section 10.2.5 of the LA (USEC, 2006c), the applicant states that the small amounts of steel scrap and other materials can be recovered and sold at market prices.

During the waste disposal process, radioactive waste would be disposed of in licensed low-level radioactive waste disposal facilities. Section 10.2.5, “Salvage and Sale,” of the LA (USEC, 2006c) estimates that approximately 60,000 cubic feet of radioactive waste would be generated during the decommissioning operation. This waste may be subject to further volume reduction before disposal. Hazardous waste will be disposed of in permitted hazardous waste disposal facilities. Non-hazardous waste and non-radioactive waste will be disposed of in a manner consistent with good industrial practice and in accordance with applicable regulations. A more complete estimate of the waste and effluents that would be generated during decommissioning will be provided to NRC in the applicant’s Decommissioning Plan. Classified components and documents will be dispositioned in accordance with the ACP Security program.

Section 10.2.7, “Final Status Survey,” of the LA (USEC, 2006c) states that the last decommissioning step is to conduct a final radiation survey to verify that the facility meets regulatory requirements for decommissioning. This survey would be compared to an initial radiation survey performed before operation in order to subtract the background radiation of the area and thus be able to identify the contribution attributable to facility operations. This final radiation survey would be documented in a report, the results would be analyzed, and further action would be taken in the decommissioning process, if required.

10.3.1.3 Management/Organization

Section 10.3, “Management/Organization,” of the LA (USEC, 2006c) states that management of the decommissioning program would assure proper training and procedures are provided to assure worker health and safety. The programs would focus on minimizing waste volumes and worker exposure to hazardous or radioactive materials. Qualified contractors assisting with decommissioning would be subject to ACP security and training requirements and procedural controls.

10.3.1.4 Health and Safety

Section 10.4, “Health and Safety,” of the LA (USEC, 2006c) states that its policy during decontamination is to keep individual and collective occupational radiation exposures as low as reasonably achievable (ALARA). The applicant’s radiation protection program would identify and control sources of radiation, establish worker protection requirements, and direct the use of survey and monitoring instruments, in accordance with ALARA principles.

10.3.1.5 Waste Management

Section 10.5, “Waste Management,” of the LA (USEC, 2006c) states that radioactive and hazardous wastes produced during decommissioning would be collected, handled, and disposed of in accordance with regulations applicable to the ACP at the time of decommissioning. Generally, procedures would be similar to those described for wastes produced during ACP operation. The applicant states that it will ultimately dispose of these wastes in licensed radioactive or hazardous waste disposal facilities located elsewhere. Non-hazardous and non-radioactive wastes would be disposed of consistent with good industrial practice and in accordance with applicable regulations.

10.3.1.6 Security and Nuclear Material Control

Section 10.6, “Security and Nuclear Material Control,” of the LA (USEC, 2006c) states that the applicant will maintain the requirements for physical security and for nuclear material control and accountability during decommissioning in a manner similar to the programs in force during ACP operation. This includes requirements for control of classified information and classified equipment described in the ACP Physical Security Program (USEC, 2006d) and the requirements for control of nuclear materials in the Fundamental Nuclear Material Control Plan (USEC, 2006b). The applicant would provide any necessary revisions to these programs in the Decommissioning Plan to be submitted near the end of plant life.

10.3.1.7 Recordkeeping

Section 10.7, “Recordkeeping,” of the LA (USEC, 2006c) states that the applicant will maintain records important for safe and effective decommissioning of the ACP in accordance with the stated procedural requirements. Information maintained in the records include:

- Records of spills or other unusual occurrences related to spread of contamination;
- As-built drawings and modifications of structures and equipment in areas where radioactive materials are used or stored, including locations that possibly could be inaccessible;
- A list contained in a single document to be updated every 2 years that will include information about recordkeeping of spills and other unusual occurrences, as required by 10 CFR 70.25 (g)(1), on certain areas within and outside the restricted areas of the facility, as specified in 10 CFR 70.25 (g)(3)(i) through (iv) and in 10 CFR Part 20; and
- Records of the cost estimates performed for the DFP, and the funding method used for assuring funds, in accordance with established records management and document control procedural requirements.

10.3.1.8 Decontamination Process

Section 10.8, “Decontamination,” of the LA (USEC, 2006c) describes the facilities, procedures, and expected results of decontamination for the ACP. Both wet and dry decontamination methods are described. Table 10.2-1 of the LA (USEC, 2006c) provides a list of components and structures for potential decontamination at decommissioning, including an estimated quantity for each component or structure.

At the time of decommissioning, NRC staff will review and evaluate the applicant’s proposed decontamination activities in the final decommissioning plan. NRC staff will also review the facility and site characterization data.

10.3.1.8.1 Facilities

Section 10.8.1, “Decontamination Service Area,” of the LA (USEC, 2006c) designates the centrifuge assembly area within X-7725 as the Decontamination Service Area, in which decommissioning activities would be conducted. This space is needed for handling centrifuges, feed, withdrawal, sampling, and transfer equipment to be disassembled and dispositioned, along with the UF_6 vacuum pumps, valves, piping, and other miscellaneous equipment. Unusable material would be destroyed. The specialized area would have a disassembly area, a buffer stock area, a decontamination area, and a scrap storage area. Equipment that may be found in the decontamination area includes transport and manipulation equipment, dismantling area, cutting machines, dismantling boxes and tanks, degreasers, citric acid and demineralized water baths, contamination monitors, wet blast cabinets, crushers or size reduction equipment, shredders, and scrubbers. The applicant states that a decontamination facility is not needed during ACP normal operation.

10.3.1.8.2 Procedures

Section 10.8.2, “Procedures,” of the LA (USEC, 2006c) describes the procedures for

decontamination. These procedures will be developed and approved by facility management to minimize worker exposure and waste volumes, and to assure work is carried out in a safe manner. At the end of useful facility life, some of the equipment, most of the buildings, and the outdoor areas should already be acceptable for release for unrestricted use in accordance with 10 CFR 20.1402. The applicant has stated that if some of the buildings, equipment, or outside areas were inadvertently contaminated during ACP operation, they would likely be cleaned up when the contamination was discovered. This limits the scope of necessary decontamination at the time of decommissioning.

The procedures that would be performed to decontaminate the centrifuges include the following: (a) removal of external fittings; (b) removal of bottom flange, motor and bearings, and collection of contaminated oil; (c) removal of top flange and withdrawal and disassembly of internals; (d) degreasing of items, as required; and (e) destruction of classified parts by methods such as shredding, crushing, and burial.

10.3.1.8.3 Results

Section 10.8.3, "Results," of the LA (USEC, 2006c) states that recoverable items would be externally decontaminated and suitable for reuse except for a very small amount of internally contaminated items where recovery and reuse is not feasible. There is potentially a small amount of salvageable scrap material. Material requiring disposal will primarily be process piping, trash, and residue from the effluent treatment systems. The applicant does not anticipate any problems that will prevent the ACP facilities from being released for unrestricted use.

Section 10.9, "Agreements with Outside Organizations," of the LA (USEC, 2006c) states that the decommissioning plan and the funding arrangements described in Section 10.10 of the LA (USEC, 2006c) and in the DFP (USEC, 2006a) provide for decontamination of the ACP for unrestricted use. Because of this, the applicant states that no agreements with outside organizations are required for control of access to the facility after shutdown and decommissioning.

10.3.1.9 Decommissioning Program Overview Summary

Based on the staff's review of the Decommissioning program overview, the staff has concluded the Decommissioning Program Overview Summary to be consistent with the guidance in Section 10.1 of NUREG-1520 (NRC, 2002a). The overview addressed the following areas:

- General decommissioning program intended to release the facility for unrestricted use;
- Features to minimize worker exposure and waste volumes;
- Planned decommissioning steps;
- Management and organization during decommissioning;
- Application of health and safety requirements;
- Management of radioactive and hazardous wastes;

- Security and nuclear material control;
- Proposed recordkeeping program to meet the requirements in 10 CFR 30.35(g), 10 CFR 40.36(f), and 10 CFR 70.25(g); and
- Decommissioning processes that identify the facilities to be used for decontamination activities, the decontamination procedures that will be prepared, and the expected decontamination results.

10.3.2 Decommissioning Costs and Financial Assurance

10.3.2.1 Decommissioning Costs

The applicant submitted decommissioning cost information consistent with the recommendations in NUREG-1757, Volume 3, "Consolidated NMSS Decommissioning Guidance - Financial Assurance, Recordkeeping, and Timeliness" (NRC, 2003). The applicant presented its decommissioning cost estimate breakdown in Appendices C and D of its DFP (USEC, 2006a). Decommissioning cost information included estimates of 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 the applicant from the transition of the Portsmouth Gaseous Diffusion Plant to a cold-standby status, as well as on the estimated cost of future tails disposition at a U.S. Department of Energy (DOE) facility.

The applicant estimates the cost of decommissioning the enrichment facility to be approximately $317.7 million in 2006 dollars, including an estimated $45.2 million for decontamination and dismantling of radioactive facility components, an estimated $50.3 million for packaging, shipping, and waste disposal, plus a 25 percent contingency. The applicant estimates the cost for tails disposition at $1,036 million in 2006 dollars, including a 25 percent contingency. The estimated total decommissioning cost, including tails disposal, is $1.35 billion.

10.3.2.1.1 Facility Decontamination and Decommissioning Costs

The cost analysis for decommissioning the centrifuges and equipment supporting the Process Building is marked as proprietary. The staff reviewed the number of centrifuges, the estimated man-hours to decontaminate the centrifuges, and the estimated volume of material resulting from the disposal of the centrifuges and supporting equipment; confirmed that the estimate includes a 25 percent contingency; and also confirmed that no credit is taken for salvage of materials or equipment. The use of a 25 percent contingency factor and taking no credit for salvage value are consistent with staff guidance in NUREG-1757, Volume 3 (NRC, 2003).

The non-proprietary estimate for decommissioning the remainder of the facility was reviewed considering: (a) labor costs; (b) decontamination methods and unit costs; (c) waste disposal costs; (d) depleted uranium disposition costs; and (e) final survey costs. Both the proprietary and non-proprietary estimates were evaluated and found to be reasonable with respect to information provided in NUREG/CR-6477, "Revised Analyses of Decommissioning Reference Non-Fuel-Cycle Facilities" (NRC, 2002b). Although the cases in NUREG/CR-6477 (NRC, 2002b) do not include a case for decommissioning a gas centrifuge enrichment plant, some

information can be compared with the cost estimates provided by the applicant. The following other considerations were also included in comparing the applicant's estimates with those in NUREG/CR-6477 (NRC, 2002b):

1. The proposed facility operates at subatmospheric pressures that minimize the spread of contamination throughout the plant;

2. The plant has design features and operating procedures to minimize releases of UF_6 (e.g., cylinder feed, withdrawal, blending, and sampling systems). These design and operating features are principally intended to minimize worker chemical exposures, but also result in low contamination levels in occupied areas of the plant;

3. Feed material will be restricted to material specifications meeting the requirements of American Society for Testing and Materials (ASTM) C787-03, "Standard Specification for Uranium Hexafluoride for Enrichment" (ASTM, 2003). Through this specification, the applicant will control the entry of other radioactive contaminants into the process systems; and

4. A significant portion of the process equipment in the plant is aluminum, which can be more easily cut and processed than the steel components assumed in the cases described in NUREG/CR-6477 (NRC, 2002b).

Based on the staff's review of the proprietary and non-proprietary information, the staff found the cost estimate for decommissioning the facility to be in a reasonable range, and the cost estimate fulfills the requirements of 10 CFR 70.25(e).

10.3.2.1.2 Depleted Uranium Disposition Costs

Under Section 3113 of the USEC Privatization Act of 1996 (Title 42 U.S. Code 2297h), DOE, "at the request of the generator, shall accept for disposal low-level radioactive waste, including depleted uranium, if it is ultimately determined to be low-level radioactive waste, generated by any person licensed by the Nuclear Regulatory Commission to operate a uranium enrichment facility." In addition, the generator must reimburse DOE for the disposal of depleted uranium in an amount equal to DOE's costs, including a pro rata share of any capital costs. On January 18, 2005, the Commission issued an order stating that depleted uranium was a low-level radioactive waste (NRC, 2005). Therefore, if the applicant requests, DOE is required under the USEC Privatization Act of 1996 to accept the depleted uranium generated by the applicant. At the request of the applicant, DOE provided a cost estimate for dispositioning depleted uranium generated by the applicant (DOE, 2005).

The applicant estimated that the facility will generate 265,300 MT of DUF_6 over a nominal 30 years of operation. The applicant estimated the waste processing and disposal cost of UF_6 tails at $4.62 per kilogram of uranium (kg U). This cost is based on the total of the 4 cost components that make up the total disposition cost for DUF_6 (i.e., deconversion, disposal, and transportation).

The disposition cost was based on the estimate from DOE providing a cost for DUF_6 disposition services as calculated by a DOE contractor (DOE, 2005) and modified by the applicant to account for the amount of depleted uranium to be generated by the applicant and in 2006

dollars (USEC, 2006a). To make the modifications, the applicant used the same method in developing disposition costs as used by the DOE contractor in preparing the original estimate for another uranium enrichment facility (DOE, 2004, 2006).

Based on the applicant's analysis, the cost estimate for dispositioning depleted uranium in 2004 dollars would be \$2.33/kg UF_6 for deconversion, \$0.003/kg UF_6 for storage, \$0.37/kg UF_6 for byproduct disposal and transportation to the licensed disposal site, \$0.17/kg UF_6 decommissioning of the deconversion plant, and \$0.09/kg UF_6 for a Federal administrative charge. Because the deconversion was assumed to take place at Portsmouth, no transportation charge from the applicant's facility to the DOE deconversion plant would be necessary. The total amount for depleted uranium disposition would be \$2.96/kg UF_6 or \$4.38/kg U in 2004 dollars and \$3.12/kg UF_6 or \$4.62/kg U in 2006 dollars. Year 2004 costs were escalated using the Implicit Price Deflator for 2005 of 2.8 percent and the administration's June 8, 2006 estimate of inflation for 2006, as measured by a forecast of the gross national product index of 2.9 percent. The total cost in 2006 dollars for dispositioning of the 265,300 MT of UF_6 estimated to be generated over the lifetime of the ACP would be \$829 million. In addition, the applicant added a 25 percent contingency factor for a total depleted uranium dispositioning cost estimate of \$1,036 million.

In the DOE reports (DOE, 2004, 2006) used as a basis for DOE dispositioning cost estimate, DOE considered 6 cost scenarios, as follows:

Scenario 1: DOE processes the applicant's depleted uranium along with its own depleted uranium concurrently at the deconversion facility in Paducah, Kentucky;

Scenario 2: DOE processes the applicant's depleted uranium along with its own depleted uranium concurrently at the deconversion facility in Portsmouth, Ohio;

Scenario 3: DOE processes its own depleted uranium first and then the applicant's depleted uranium at the deconversion facility in Paducah, Kentucky;

Scenario 4: DOE processes its own depleted uranium first and then the applicant's depleted uranium at the deconversion facility in Portsmouth, Ohio;

Scenario 5: DOE adds an additional deconversion line at the deconversion plant in Paducah, Kentucky, to increase its total annual capacity from 18,000 metric tons of UF_6 to 24,750 metric tons of UF_6; and

Scenario 6: DOE adds an additional deconversion line at the deconversion plant in Portsmouth, Ohio, to increase its total annual capacity from 13,500 metric tons of UF_6 to 20,250 metric tons of UF_6.

For each of the above scenarios, the depleted uranium disposition costs were computed and the highest cost scenarios (Scenarios 2 and 4) were selected as the basis for its cost estimate. Interestingly, Scenarios 1 and 3 had lower costs, even with the additional costs of transportation from Portsmouth to Paducah, because of the lower costs associated with higher production rates. The applicant also evaluated the costs for uranium disposition if the Portsmouth deconversion facility processed not only the planned DOE depleted uranium and the depleted uranium from the ACP, but also the depleted uranium from an additional uranium enrichment

facility. For this scenario, the costs were lower than the costs from Scenarios 2 and 4 above.

The staff reviewed the above information to determine if all appropriate costs were considered and the information was adequately explained. The staff reviewed the cost estimates for: (a) the deconversion (pro rata capital and operation costs) of the depleted uranium tails at a DOE facility; (b) transportation of the byproducts of the tails conversion (depleted uranium oxides and calcium fluoride) to a licensed low-level radioactive waste disposal facility; (c) storage of tails before disposal; (d) disposal; and (e) pro rata decontamination and decommissioning costs for the conversion facility. The deconversion costs were based on contractual cost information provided by the DOE deconversion contractor, which was awarded the contract to construct and operate the deconversion facilities in Portsmouth and Paducah.

The operation costs include costs for deconversion operations, cylinder management disposal activities (including waste preparation and characterization, and transportation to and disposal of depleted uranium oxides and calcium fluoride at a low-level radioactive waste disposal facility, plant management and administration, and a management reserve and fee). The pro rata share of the capital costs were determined by allocating the total proposed baseline construction costs as a ratio of UF_6 to be processed from the applicant and from DOE.

The cost estimate properly does not assume any resale or reuse of products resulting from the conversion process. Since DOE plans to use the incoming UF_6 cylinders as disposal containers for the depleted uranium oxides generated by deconversion, the costs to prepare the cylinders as disposal containers are included in the operational costs of the cost estimate. Since construction of the deconversion facilities will be performed under a DOE contract, where the contractor will be reimbursed for construction costs as they occur, the contractor will have no need to take out construction loans and account for debt service and cost of capital.

The transportation and disposal costs also were based on the estimates for those costs in the DOE letters (DOE, 2005, 2006), which were in turn based on information concerning the costs of services provided by transportation and disposal service vendors. The transportation estimate is based on shipment of the products of the deconversion process, depleted uranium oxides and calcium fluoride, to a licensed low-level radioactive waste disposal facility by rail. Because the deconversion was assumed to take place at Portsmouth no transportation charge from the applicant's facility to the DOE deconversion plant would be necessary. The cost of loading cylinders at the applicant's facility for transfer to DOE is considered to be negligible.

The applicant stated in Section 3.0 of the DFP (USEC, 2006a) that it believed the depleted uranium dispositioning cost estimate should be viewed as a conservative value based on a number of factors and assumptions. Those factors included, in the applicant's opinion: (a) location(s) for processing the applicant's depleted uranium; (b) transportation costs; (c) escalation rate(s) of various construction cost components; (d) de-escalation rate(s) of future operating costs (to present day dollars); (e) volume of tails disposed; (f) revenue/avoided cost from sale of conversion products (e.g. hydrogen fluoride) or higher assay tails (tails stripping); (g) construction and operations budget contingencies; (h) allocation of decontamination; and (i) decommissioning costs (between the applicant and DOE), and DOE oversight costs. The staff confirmed, however, that no credits for salvage value (e.g. from sale of conversion products or higher assay tails) were included in the decommissioning cost estimate.

10.3.2.1.3 Decommissioning Cost Summary

Based on the staff's review of the proprietary and non-proprietary information, the staff found the cost estimate for decommissioning the facility to be in a reasonable range, and the cost estimate fulfills the requirements of 10 CFR 70.25(e) and the evaluation criteria in Section 4.1 of NUREG-1757, Volume 3 (NRC, 2003) for the following reasons:

- The cost estimate is based on documented and reasonable assumptions;
- The cost estimates for individual facility activities and components are reasonable and, to the extent possible, consistent with NRC cost estimation reference documents;
- The cost estimate reflects decommissioning under appropriate facility conditions;
- The cost estimate includes costs for labor, equipment and supplies, overhead and contractor profit, sampling, and miscellaneous expenses;
- The cost estimate includes costs for all major decommissioning activities, including planning and preparation; decontamination or dismantling facility components; packaging, shipping, and disposal of wastes; restoration of facility grounds; and the final radiation survey;
- The computations are correct;
- No credit is taken for salvage value;
- The decommissioning cost estimate includes an adequate contingency factor of 25 percent;
- The decommissioning cost estimate provides a description of how it will be adjusted periodically over the life of the facility; and
- The depleted uranium disposition cost estimate is based on information from DOE, which is statutorily required to recover costs from a uranium enrichment facility licensee if the licensee chooses to utilize a DOE disposition path for depleted uranium generated at the facility.

10.3.2.2 Financial Assurance for Decommissioning

The applicant stated it will use a surety bond method to provide reasonable assurance of decommissioning funding as required by 10 CFR 70.25(f)(2). The applicant provided draft copies of the surety bond and standby trust language. The applicant stated, however, that it may choose to use alternate financial assurance funding methods. Finalization of the specific financial instruments to be used will be completed, and signed originals of those instruments will be provided to NRC for final confirmation of the instrument before the applicant receiving licensed material at the facility. In addition, the applicant committed to provide continuous financial assurance through the completion of decommissioning and termination of the licenses.

The staff reviewed the surety bond method to be adopted by the applicant and concluded that it 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 (NRC, 2003). Accordingly, the applicant stated that its surety bond will contain, but not be limited to, the following attributes:

- The surety bond will be open-ended or, if written for a specified term, such as 5 years, will be renewed automatically unless 90 days or more before the renewal date, the issuer notifies NRC, the trust to which the surety is payable, and the applicant of its intention not to renew. The surety bond will also provide that the full face amounts are paid to the beneficiary automatically before the expiration without proof of forfeiture if the applicant fails to provide a replacement acceptable to NRC within 30 days after receipt of notification of cancellation;

- The surety bond will be payable to a standby trust established for decommissioning costs. The trustee and trust will be ones acceptable to NRC. For instance, the trustee may be an appropriate State or Federal government agency or an entity which has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency; and

- The surety bond and standby trust will remain in effect until NRC has terminated the license.

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 the following license condition:

> *The licensee shall provide final copies of the proposed financial assurance instruments to NRC for review at least six months prior to the planned date for obtaining licensed material, and provide to NRC final executed copies of the reviewed financial assurance instruments prior to the receipt of licensed material. The amount of the financial assurance instrument shall be updated to current year dollars and include any applicable changes to the decommissioning cost estimate. The decommissioning cost estimate shall include an update to USEC's Analysis of Depleted Uranium Disposal Costs for the ACP. To develop this update, USEC shall coordinate with DOE to determine necessary changes to the DOE contractor's depleted uranium cost estimate utilized as input to the USEC specific analysis.*

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 operation of additional separations building modules and accumulated tails. The applicant committed to providing such updates annually on a forward-looking incremental basis. In addition, as discussed in Sections 1.2.5 and 10.10.4 of the LA (USEC, 2006c), the applicant requested an exemption from the 10 CFR 40.36 and 10 CFR 70.25 decommissioning funding requirements to allow incremental funding for decommissioning based on the expected number of centrifuges to be built and installed and on the expected amount of depleted uranium tails to be generated annually in a forward-looking manner. As discussed in Section 10.10.4 of the LA (USEC, 2006c), the

applicant stated that, up to the time of full capacity operation, it would initially provide funding for the projected cost of facility decontamination and decommissioning, assuming operation at full capacity, except for the following:

1. Decontamination and removal of the centrifuges are incrementally funded on an annual forward-looking basis; and

2. The UF_6 tails are funded as they are generated on an annually forward-looking basis.

Once full capacity operation is achieved, the applicant proposed to update the cost estimate for depleted uranium byproduct generation on an annual forward-looking basis and update the cost estimates for decontamination and decommissioning the remainder of the facility at intervals not to exceed 3 years.

As stated in Section 1.2.3.6 of this SER, 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 is granting the requested exemption.

NRC staff will review the initial cost estimate and the expected financial instrument before the applicant takes possession of licensed material. NRC staff will also review all subsequent annual revisions to the cost estimate and financial instruments. NRC staff will include the following license condition related to the submittal of DFP updates:

> *The initial and subsequent updated DFP cost estimates, up to the time of full capacity operations, and revised funding instruments shall be provided annually and shall provide full funding for decontamination and decommissioning of the full-size facility, except*
>
> *(1) The cost estimate for decontamination and removal of the centrifuges shall be provided on an annual forward-looking basis based on planned incremental enrichment capacity increases; and*
>
> *(2) The cost estimate for depleted uranium byproduct generation shall be provided on a projected annual forward-looking basis. The decommissioning cost estimate shall include an update to USEC's Analysis of Depleted Uranium Disposal Costs for the ACP. To develop this update, USEC shall coordinate with DOE to determine necessary changes to the DOE contractor's depleted uranium cost estimate utilized as input to the USEC specific analysis.*
>
> *Once full capacity operation is achieved, the licensee shall provide cost estimates for depleted uranium byproduct generation on an annual forward-looking basis and cost estimates for decontamination and decommissioning the remainder of the facility at intervals not to exceed 3 years, consistent with the requirements of 10 CFR 40.36(d) and 10 CFR 70.25(e).*

The DFP cost estimates shall be provided to NRC for review, and subsequently, after resolution of any NRC comments, final executed copies of the financial assurance instruments shall be provided to NRC.

10.4 EVALUATION FINDINGS

NRC staff has evaluated the applicant's plan for decommissioning activities, its cost estimates for those activities, and its proposal to provide a financial assurance mechanism for decommissioning in accordance with NUREG-1757, Volume 3 (NRC, 2003). In addition, as noted above, the staff has included a license condition to address the applicant's commitments for submitting the initial executed financial assurance instruments and for updating the DFP over time. On the basis of this evaluation, NRC staff has determined that the applicant has considered site-specific decommissioning activities that may be needed in the future. In addition, the applicant's plan for decommissioning, including financial assurance for decommissioning, complies with NRC's regulatory requirements in 10 CFR Parts 20, 30, 40, and 70 as identified in Section 10.1 above in this SER; provides sufficient funding to ensure decommissioning and decontamination of the facility, even if the applicant is unable to meet its financial obligations; and provides for updating the decommissioning cost estimate for the ACP and the associated funding levels over the life of the facility. Therefore, NRC staff finds that the applicant's plan for decommissioning activities provides reasonable assurance of protecting the health and safety of workers, the public, and the environment.

However, 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 the following license conditions:

The licensee shall provide final copies of the proposed financial assurance instruments to NRC for review at least six months prior to the planned date for obtaining licensed material, and provide to NRC final executed copies of the reviewed financial assurance instruments prior to the receipt of licensed material. The amount of the financial assurance instrument shall be updated to current year dollars and include any applicable changes to the decommissioning cost estimate. The decommissioning cost estimate shall include an update to USEC's Analysis of Depleted Uranium Disposal Costs for the ACP. To develop this update, USEC shall coordinate with DOE to determine necessary changes to the DOE contractor's depleted uranium cost estimate utilized as input to the USEC specific analysis.

In addition, NRC staff is imposing the following license condition for updating DFP:

The initial and subsequent updated DFP cost estimates, up to the time of full capacity operations, and revised funding instruments shall be provided annually and shall provide full funding for decontamination and decommissioning of the full-size facility, except

(1) The cost estimate for decontamination and removal of the centrifuges shall be provided on an annual forward-looking basis based on planned incremental enrichment capacity increases; and

(2) The cost estimate for depleted uranium byproduct generation shall be provided on a projected annual forward-looking basis. The decommissioning cost estimate shall include an update to USEC's Analysis of Depleted Uranium Disposal Costs for the ACP. To develop this update, USEC shall coordinate with DOE to determine necessary changes to the DOE contractor's depleted uranium cost estimate utilized as input to the USEC specific analysis.

Once full capacity operation is achieved, the licensee shall provide cost estimates for depleted uranium byproduct generation on an annual forward-looking basis and cost estimates for decontamination and decommissioning the remainder of the facility at intervals not to exceed 3 years, consistent with the requirements of 10 CFR 40.36(d) and 10 CFR 70.25(e).

The DFP cost estimates shall be provided to NRC for review, and subsequently, after resolution of any NRC comments, final executed copies of the financial assurance instruments shall be provided to NRC.

10.5 REFERENCES

(ASTM, 2003) American Standard for Testing Materials (ASTM). ASTM C787-03, "Standard Specification for Uranium Hexafluoride Enrichment," 2003.

(DOE, 2004) U.S. Department of Energy (DOE). DE523T1, "An Analysis of DOE's Cost to Dispose of DUF_6 (Depleted Uranium Hexafluoride)," December, 2004.

(DOE, 2005) U.S. Department of Energy (DOE). "Conversion and Disposal of Depleted Uranium Hexafluoride (DUF_6) Generated by USEC at the American Centrifuge Plant in Piketon, Ohio," December 12, 2005.

(DOE, 2006) U.S. Department of Energy (DOE). "Conversion and Disposal of Depleted Uranium Hexafluoride (DUF_6) Generated by USEC at the American Centrifuge Plant in Piketon, Ohio," February 10, 2006.

(NRC, 2000) U.S. Nuclear Regulatory Commission (NRC). NUREG-1727, "NMSS Decommissioning Standard Review Plan," September, 2000.

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

(NRC, 2002b) U.S. Nuclear Regulatory Commission (NRC). NUREG/CR-6477, "Revised Analysis of Decommissioning Reference Non-Fuel-Cycle Facilities." 2002.

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(USEC, 2006a) USEC Inc. (USEC). "Decommissioning Funding Plan," Revision 9, August 2006.

(USEC, 2006b) USEC Inc. (USEC). "Fundamental Nuclear Material Control Plan for the American Centrifuge Plant in Piketon, Ohio," Revision 5, August 2006.

(USEC, 2006c) USEC Inc. (USEC). "License Application for the American Centrifuge Plant in Piketon, Ohio," Revision 18, September 2006.

(USEC, 2006d) USEC Inc. (USEC). "Physical Security Plan for the Protection of Special Nuclear Material of Low Strategic Significance of the Security Program for the American Centrifuge Plant in Piketon, Ohio," Revision 5, August 2006.

11.0 MANAGEMENT MEASURES

Management measures are functions that the applicant performs, generally on a continuing basis, which are applied to items relied on for safety (IROFS), to ensure compliance with established performance requirements and that the IROFS are available and reliable. Management measures shall be implemented to assure compliance with performance requirements and the degree to which they will be applied will be a function of the item's importance in terms of meeting performance requirements as evaluated in the Integrated Safety Analysis (ISA). This chapter addresses each of the management measures included in the 10 CFR Part 70 definition of management measures, including: (a) configuration management (CM); (b) maintenance; (c) training and qualifications; (d) procedures; (e) audits and assessments; (f) incident investigations; (g) records management; and (h) other quality assurance (QA) elements.

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's management measures is to evaluate whether the application provided information to ensure that the management measures applied to IROFS, as documented in the ISA Summary, provide adequate 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 will also determine whether the management measures are applied to the IROFS in a manner commensurate with the IROFS' importance to safety.

11.1 REGULATORY REQUIREMENTS

The requirements for fuel cycle facility management measures are specified in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

1. 10 CFR 70.4 states that management measures include: (a) CM; maintenance; (b) training and qualifications; (c) procedures; audits and assessments; (d) incident investigations; (e) records management; and (f) other QA elements.

2. 10 CFR 70.62(a)(3) states that records must be kept for all IROFS failures; describes required data to be reported; and sets time requirements for updating the records.

3. 10 CFR 70.62(d) requires an applicant to establish management measures, for application to engineered and administrative controls and control systems that are identified as IROFS, pursuant to 10 CFR 70.61(e), to ensure they are available and reliable.

4. 10 CFR 19.12 states requirements for worker instructions that are applicable to personnel training and qualifications.

5. 10 CFR 70.22(a)(8) states requirements for license applications to address proposed procedures to protect health and minimize danger to life and property.

6. 10 CFR 70.72 requires a licensee to establish a CM program to evaluate, implement, and track changes to the facility; structures, systems and components (SSCs);

processes; and activities of personnel.

7. 10 CFR 70.74(a) and (b) state requirements for incident investigation and reporting.

11.2 REGULATORY GUIDANCE AND ACCEPTANCE CRITERIA

The guidance applicable to NRC's review of the management measures description section of the license application (LA) (USEC, 2006b) is contained in Chapter 11 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). This chapter is applicable in its entirety. The acceptance criteria applicable to this review are contained in Section 11.4.3 of NUREG-1520 (NRC, 2002).

11.3 STAFF REVIEW AND ANALYSIS

NRC staff has reviewed the CM function for the American Centrifuge Plant (ACP) in accordance with the regulatory acceptance criteria of NUREG-1520 (NRC, 2002) Section 11.4.3.

11.3.1 Configuration Management Program

The applicant's CM function is described in Section 11.1 of the LA (USEC, 2006b).

11.3.1.1 CM Policy

The goal of the CM program is to assure that the ACP has accurate, current documentation that matches the plant's physical/functional configuration, while complying with applicable requirements. The CM program will be implemented to ensure that changes from the plant baseline configuration are identified and controlled to help ensure safety through consistency among the plant design and operational requirements, the physical configuration, and the plant documentation. The applicant has defined the Configuration Management Policy for the ACP and provided a program overview in Section 11.1 of the LA (USEC, 2006b). The CM program will include: (1) identification and documentation of IROFS; (2) organizational descriptions of duties and responsibilities; and (3) administrative controls, procedures, and policies to implement and document activities that maintain the facility's configuration.

The applicant states that the CM program will be applied to the plant, structures, processes, systems, equipment, components, computer programs, and activities of personnel, regardless of the item's Quality Level (QL) classification. Hence, all IROFS listed in the ISA Summary (USEC, 2006a) will be under the CM program. CM program procedures will provide for a graded application of resources taking into consideration: (a) QL (risk significance); (b) applicable regulations, (c) industry codes and standards; (d) complexity or uniqueness of an item or activity and the environment in which it has to function; (e) quality history of an item in service; (f) degree to which functional compliance can be demonstrated or assessed by test, inspection, or maintenance methods; anticipated life span; (g) degree of standardization; (h) importance of data generated; (i) reproducibility of results; and (j) consequence of failure. Section 11.3.8.2 of this Safety Evaluation Report (SER) discusses the application of Quality Levels (QLs) to IROFS listed in the ISA Summary (USEC, 2006a).

Implementing procedures will be established to provide a management system that will evaluate, implement, and track each change. The procedures will address: (a) the technical

basis for the change; (b) the impact of the change on safety and health or control of licensed material; (c) revisions, if required, to existing operating procedures, including any necessary training, or retraining before operations; authorization requirements for change; (d) for temporary changes, the approved duration of the change; and (e) the impacts or modifications to the ISA, the ISA Summary (USEC, 2006a), or other safety program information that will be part of the LA (USEC, 2006b).

The functional interface between engineering, procurement, operations, maintenance, production support and QA is described in Section 11.1.1.2 of the LA (USEC, 2006b). The Engineering Manager will be assigned primary responsibility for implementation of the CM program and is the plant Design Authority (DA). Responsibilities include: (a) establishing the design requirements; (b) ensuring design output information (documentation and data) appropriately and accurately reflects the design input; and (c) maintaining the plant's ISA and ISA Summary (USEC, 2006a). Other Engineering responsibilities include: (a) management of the temporary change process; (b) the identification and definition of IROFS as part of the ISA process; (c) review of facility changes; (d) establishment of IROFS inspection and acceptance criteria; (e) definition of IROFS boundaries; (f) the establishment and maintenance of a controlled data base for IROFS information; (g) document and procedure updates addressing modifications; and (h) work package development. The Engineering Manager will be responsible for the development and operation of the Records Management and Document Control (RMDC) program.

The Procurement Manager will be responsible for the development of procedures in accordance with the Quality Assurance Program Description (QAPD) for: (a) procurement and control of items; (b) the purchase of IROFS and replacement parts in accordance with the requirements and technical specifications identified by the engineering organization; (c) ensuring that only accepted IROFS are stored and issued for work; and (d) items are maintained in a manner that complies with Engineering issued requirements.

The Operations Manager ensures that modifications will not be made to the facility design or operational configuration without proper review and approval. In addition, the Operations Manager also ensures: (a) that pre-operational, operational, and post maintenance and post-modification tests/checks will be performed and documented to assure that IROFS will be operating as intended; (b) that maintenance, testing and modification activities will be properly authorized; and (c) that approved procedures will be used for operations involving the replacement or adjustment of IROFS.

The Maintenance Manager will be responsible for: (a) the development and implementation of the work control process; (b) ensuring that maintenance personnel are knowledgeable of the requirements for working on IROFS; (c) obtaining authorization to perform work on IROFS; and (d) ensuring that modifications will not be made to a design or operational configuration without proper review and approval.

The Production Support Manager will be responsible for development of a procedures control program to ensure technical, operations, maintenance, and administrative procedures used to apply the CM program processes are properly developed, reviewed, approved, revised, and controlled. The Production Support Manager also will be responsible for providing technical training support to plant personnel who will be relied upon to operate, maintain, or modify IROFS, and to provide training support to Engineering, Operations, and Maintenance personnel

to ensure training is updated as a result of changes to the plant.

The QA Manager will: (a) assist in the acceptance processes for commercial grade and non-commercial grade IROFS; (b) verify that the DA supplied acceptance criteria will be met for accepted items; (c) conduct appropriate audits and surveillances of processes that implement the CM program; (d) conduct in-process inspection of maintenance work packages; and (e) audit vendors and suppliers in accordance with the QAPD.

The staff finds that the applicant has provided a sufficient description of the overall CM policy and functions. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.1(1) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.1.2 Design Requirements

Section 11.1.2, "Design Requirements," of the LA (USEC, 2006b) states that design requirements will be developed for IROFS or other systems or components required to support the ISA and meet the baseline design criteria (BDC) as defined in 10 CFR 70.64. Support of environmental impact and mission-based functions will be identified in System Requirements Documents (SRDs). Design requirements will be developed by the Engineering Organization and documented in Design Criteria Documents (DCDs) for each plant system. The Engineering Manager will be the plant Design Authority and will approve the DCDs. Changes to any design basis or design requirement will be considered a modification, and will be controlled by the change control process described in Section 11.1.4 of the LA (USEC, 2006b). The baseline configuration of the facility will consist of approved DCDs and ISA Summary (USEC, 2006a), Design Basis Documents, SRDs, and the as-built drawings and specifications and will be approved by the Engineering Manager. All changes to these documents will be controlled in accordance with the RMDC requirements, described in Section 11.7 of the LA (USEC, 2006b). The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.1(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.1.3 Document Control

Section 11.1.3, "Document Control," of the LA (USEC, 2006b) states that the procedure control program, which is described in Section 11.4 of the LA (USEC, 2006b), will assure that procedures are generated, reviewed, approved, and distributed in a controlled manner. An index of documents required to support the CM program will be controlled and maintained. The RMDC program, which is described in Section 11.7 of the LA (USEC, 2006b), will assure that changes to approved and controlled documents will be issued in a timely manner, distributed to controlled copy holders, and will be maintained available to support daily work activities. Controlled documents will include, but are not limited to: (a) procedures affecting activities of IROFS; (b) design documents; (c) IROFS data base change records; (d) engineering specification data sheets; (e) the ISA Summary (USEC, 2006a) and hazard analyses; (f) assessment report; (g) emergency operating and response plans and procedures; and (h) records to support maintenance and verification of plant configuration. Further examples of records are identified in Section 11.7.5 of the LA (USEC, 2006b). The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.1(3) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.1.4 Change Control

Section 11.1.4, “Change Control,” of the LA (USEC, 2006b) states that changes to the plant, structures, systems, processes, equipment, components, computer programs, and activities of personnel, will be implemented in accordance with 10 CFR 70.72. The Plant Safety Review Committee will review changes to the facility or facility’s operations, including tests and experiments, as specified in the procedures. The procedures also specify approval authority for changes. The change control process will be implemented in accordance with approved procedures. The DA (Engineering Manager) will review requests for engineering assistance for acceptability review, engineering approval, and disposition. Before implementing any change, procedures will assure that the impacts on modifications to the ISA, ISA Summary (USEC, 2006a), or other safety program information are addressed.

Construction Project requests for plant modifications, additions, or changes will have a 10 CFR 70.72 review performed to determine whether prior NRC approval will be required. Permanent and temporary modifications will be evaluated for any required changes or additions to the plant’s procedures, personnel training, testing programs, or the ISA Summary (USEC, 2006a). The modifications will also be evaluated, as appropriate, for potential radiation exposure, potential chemical exposure, nuclear criticality safety, and worker safety requirements and/or restrictions. The design process will include identification of critical repair parts for IROFS. Proposed changes will receive an independent, technical review that identifies appropriate interfaces for inclusion in the change package (e.g., procedures, training, safety).

A final review will be conducted before release into operations. This review will verify that: (a) the safety analysis documentation will be complete and approved; (b) operational procedure changes, if required, will be completed and other supporting procedure changes have been initiated; (c) operational training and qualification changes, if required, have been completed; (d) design changes have been completed and any as-built changes are identified and approved; (e) document changes, if required, are completed; temporary change duration will be documented and the modified equipment is tagged; (f) post-modification testing has been successfully completed; and (g) all appropriate approvals have been obtained.

Changes to procedures and controlled documents will be controlled in accordance with the programs described in Sections 11.4, “Procedures,” and 11.7, “Records Management and Document Control,” of the LA (USEC, 2006b).

The staff finds that the applicant has provided an adequate description of the change control process. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.1(4) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.1.5 Assessments

Section 11.1.5, “Assessments,” of the LA (USEC, 2006b) states that the CM Assessment program includes both initial and periodic document and system walk-down assessments. An initial assessment will be performed during the applicant’s readiness review of the ACP. This is performed in order to provide field verification of design requirements and documentation and verification of procedures and training. Periodic assessments will be performed as discussed in Section 11.5 of the LA (USEC, 2006b) and the QAPD (USEC, 2006c). Deficiencies or recommendations will be documented and addressed in accordance with the Corrective Action program described in Section 11.6 of the LA (USEC, 2006b). The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.1(5) of NUREG-1520

(NRC, 2002), and is therefore acceptable.

11.3.1.6 Design Reconstitution

Section 11.1.6, “Design Verification,” of the LA (USEC, 2006b) states that the applicant will lease portions of structures that were already built by the U.S. Department of Energy for the Gas Centrifuge Enrichment Program for ACP use. The applicant intends to use existing SSCs. To verify that the design and construction of the existing SSCs meet the ACP design requirements, the applicant will utilize a verification process that will include an assessment of the SSC to compare the configuration with original drawings, construction specifications, and procedures. Where appropriate, system walk-downs will be performed. The assessment results will be evaluated to verify that the SSC fulfills the requirements established by the system requirements document. Appropriate design changes will be made where existing SSCs do not meet design requirements. SSCs will be verified or modified to meet the design requirements and will be incorporated into the plant baseline configuration information. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.1(6) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.2 Maintenance

Section 11.2, “Maintenance,” of the LA (USEC, 2006b) states that the maintenance programs related to preventive and corrective maintenance will be established to provide a level of inspection, calibration, repair, replacement, and testing to ensure each IROFS will be available and reliable to perform its intended function. Policies, procedures, and programs will address: (a) personnel qualifications and training; (b) design/work control; (c) corrective and preventive maintenance; (d) surveillance and monitoring; (e) post-maintenance testing; (f) control of measuring and test equipment; and (g) equipment/work history. The Maintenance Manager will be responsible for the overall coordination and management of the organization.

Personnel qualification and training requirements for maintenance personnel will be established that take into consideration the importance of the activities performed under each job classification to the safe operation of the facility and maintenance of IROFS, the complexity of the activity, the frequency of the task, and the consequences of performance errors. Consideration will also be given to skill-of-the-craft and availability of procedures. Contractors performing work on IROFS will be given the same guidance under the oversight of the ACP organization.

The applicant has stated that maintenance of plant equipment will be performed in a manner that maintains the documented configuration of plant systems. A work control process will be established to provide for the control, review, and approval process in order to maintain the documented configuration of ACP systems. The level of maintenance planning, the extent of reviews, and the approval required to perform the maintenance task will be graded based on the QL of the item. Before removing an IROFS from service, appropriate compensatory measures will be established. The repair and replacement of IROFS will be performed with like-for-like parts or substitute parts approved by the Engineering Organization. Modifications will only be performed following evaluation and approval of the Engineering Organization. Review and approval by the equipment owner and maintenance will be required before a work package can be used to perform maintenance on plant equipment. The Engineering Organization will be required to review and approve work packages created for maintenance of QL-1 and QL-2 items

and packages developed for modification of plant systems. The Operations Organization will be required to authorize the performance of maintenance and removal of IROFS from service. The overall work control process provides configuration control of ACP equipment.

11.3.2.1 Corrective Maintenance

Section 11.2.4, "Corrective Maintenance," of the LA (USEC, 2006b) states that the corrective maintenance process will address those actions to check, troubleshoot, and repair equipment that has degraded or failed. The identification, prioritization, planning, and scheduling of corrective maintenance activities will be accomplished in accordance with the work control process described in Section 11.2.3 of the LA (USEC, 2006b). Corrective maintenance will be performed to remediate unacceptable performance deficiencies in an IROFS and to eliminate or minimize the recurrence of these deficiencies. After corrective maintenance on an item, a functional test will be performed to confirm that the maintenance performed was satisfactory, the deficiency has been corrected, and the maintenance activity did not adversely affect the reliability of the item. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.2(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.2.2 Preventive Maintenance

Section 11.2.5, "Preventive Maintenance," of the LA (USEC, 2006b) states that preventive maintenance (PM) will be performed on a periodic basis to prevent failures, facilitate performance, and maintain or extend the life of equipment. PM will help ensure that QL items will be available to perform their function and will be reliable. PM tasks will be developed through a review of manufacturer recommendations, industry standards, and available historical operating information. The rational for deviations from manufacturer recommendations or industry standards will be documented. The formal documented bases for the tasks will be developed, evaluated, and approved by the Engineering Organization. Changes to the tasks may be made to modify the frequency or scope of PM work as a result of feedback, corrective maintenance, and incident investigations. After preventive maintenance on an item, a functional test will be performed to confirm that the maintenance performed was satisfactory and the maintenance activity did not adversely affect the reliability of the item. Section 11.2.9, "Equipment/Work History," of the LA (USEC, 2006b) states that maintenance of an IROFS requires the preparation of a work package that contains an equipment history form to collect information during preventive and corrective maintenance on IROFS. This data is used to determine whether modifications to the maintenance program is necessary and ensures that IROFS are available and reliable. The actual documentation is controlled according to the RMDC program practices. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.2(3) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.2.3 Surveillance and Monitoring

Section 11.2.6, "Surveillance and Monitoring," of the LA (USEC, 2006b) states that surveillance and monitoring will be performed at specified intervals to verify the proper operation of IROFS and to measure the degree to which IROFS meet performance requirements. The surveillances consist of performance checks, calibrations, tests, and/or inspections, and will be performed for IROFS at specified intervals to verify that their design functions are met. The established frequencies will be determined by the IROFS' degree of safety importance. The surveillance

program will adhere to “Inspection, Testing, and Maintenance Baseline Design Criteria,” as stated in 10 CFR 70.64. The surveillances will be included in the work control process to permit timely planning, scheduling, establishment of system or plant conditions, execution of the activity, and documentation that identifies the results of the surveillance. The results of surveillance activities will be trended. Frequencies will be adjusted or other corrective actions taken as necessary. Activities involving incident investigation and IROFS failures, as well as records of IROFS failures, are discussed in Section 11.3.6 of this SER. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.2(1) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.2.4 Functional Testing

Section 2.3, “Pre-operational Testing and Initial Start-up,” of the LA (USEC, 2006b), describes the plans to ensure safe turnover, testing, and start-up of the centrifuge machines, equipment, and support systems. The plans will cover the transition from the refurbishment and construction phases to the operations phase. The Engineering Manager will be responsible for the development and implementation of testing to provide for the turnover and acceptance of equipment and systems from contractors and vendors. The Operations Manager, with assistance from the Engineering Manager, will be responsible for the development and execution of the Integrated Systems and associated Test Plans, which will be used to demonstrate proper operation of systems and ensure that the systems meet their intended design functions. Testing documentation will be maintained in accordance with the RMDC requirements.

The Engineering Manager will be responsible for coordination of turnover and acceptance testing after the refurbishment and construction contractor(s) have completed: (a) as-built drawing verification; (b) purging and flushing, cleaning; (c) hydrostatic testing; (d) system turnover; (e) initial calibration of instrumentation in accordance with procedures; (f) design documents; and (g) installation specifications. Integrated systems testing will then be performed as required by pertinent design codes or the QAPD that were not performed by the contractor before turnover. The testing will principally be associated with IROFS and will demonstrate that the IROFS are capable of performing their intended function. The testing will generally occur before the introduction of licensed material. Pre-operational tests not required before the introduction of licensed material may be performed following introduction of the licensed material into the process.

The objectives of the pre-operational testing program are to ensure that the facilities and systems have been adequately designed and constructed; that they meet regulatory and licensing requirements; they do not adversely affect worker or public health and safety; and the systems can perform their intended functions.

The purpose of the initial start-up testing is to ensure that structures, systems, and components will perform their intended design functions in a safe manner. Examples of initial start-up tests include leak testing, evacuation, start-up, and filling of centrifuges.

The applicant states in Section 11.2.2 of the LA (USEC, 2006b) that contractors that work on or are performing activities that could affect IROFS follow the same maintenance guidelines as maintenance personnel. A member of the ACP organization will provide oversight for contract

activities.

Once turnover is complete, Section 11.2.7, “Functional Testing,” of the LA (USEC, 2006b) states that a post-maintenance testing program will be established to provide assurance that QL items will perform their intended safety function following maintenance. The test requirements will be included in work packages and the test results will be documented. The testing will confirm that the maintenance was satisfactorily performed, any identified deficiency corrected, and that the reliability of the QL item was not adversely affected.

To assure the required accuracy, range, and stability, a measuring and test equipment (M&TE) program will be developed to assure that M&TE will be properly controlled, calibrated, and adjusted at specified time periods. Available standards, traceable to the National Institute of Standards and Technology will be used. Where standards will not be available, the Engineering Organization will approve and document the bases for calibration.

The work control process will collect data from the performance of preventive and corrective maintenance activities in order to identify the need for modifications and improvements for the maintenance program and to improve the reliability of IROFS. A database will be maintained for historic purposes and used by the Engineering Organization to evaluate the reliability of IROFS.

Records showing the functional test schedule and results of all IROFS in accordance with 10 CFR 70.64(a) requirements will be maintained under the provisions of the Records Management program described in Section 11.7.1 of the LA (USEC, 2006b).

Administrative controls are often identified as IROFS. Section 11.4.7 of the LA (USEC, 2006b) states that in-hand procedures, check sheets, or other approved operator aids are developed for IROFS that have extensive or complex tasks, tasks infrequently performed, or tasks in which operations must be performed in a specified sequence. Reference use procedures are provided for other tasks and are not required to be present in the work area. The applicant will ensure that personnel are trained on the use of procedures and are trained and qualified.

The work control process will establish the necessary control, review, and approval process to maintain the documented configuration of plant systems. All maintenance activities under the work control process require an evaluation of availability of qualified personnel to perform the maintenance, approved work instructions and/or procedures, approved parts or substitutes, drawings, and safety permits. Operations is required to authorize maintenance performance and removal of IROFS from service. Operations is also responsible for ensuring safe operations during removal of IROFS from service, including establishing compensatory measures. Operations will be notified upon completion of these activities.

The staff finds that the applicant has provided an adequate description of functional testing. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.2(4) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3 Training and Qualifications

The applicant discusses training and qualification programs in Section 11.3 of the LA (USEC,

2006b). The programs will ensure that those personnel who perform activities relied on for safety have the applicable knowledge and skills necessary to design, operate, and maintain the plant in a safe manner. Performance Based Training methodology will be used for those tasks associated with the design, modification, operation, or maintenance of IROFS identified in the ISA Summary (USEC, 2006a). Personnel will be trained and tested as necessary in order to ensure that personnel will be qualified on practices important to public and worker safety, safeguard of licensed material, and protection of the environment.

11.3.3.1 Organization and Management of the Training Function

In Section 11.3.1, "Organization and Management of the Training Function," of the LA (USEC, 2006b), the applicant states that the Training Manager will be responsible for establishing procedures governing the application of the performance base training methodology for the analysis, design, development, implementation and evaluation of the training programs. Training personnel will interface with the line managers for training development and implementation. The functional organization managers will be responsible for defining the job-specific training needs and ensuring completion of training and qualification for personnel within their organization. Training attendance will be tracked and work restrictions will be placed on employees who have training deficiencies.

Workers relied upon to design, operate, or maintain IROFS will be trained and evaluated for their qualifications before the assignment of these duties. Initial training contains classroom and on-the-job (OJT) training necessary to provide an understanding of the fundamentals, basic principles, systems, procedures, and emergency responses associated with the employee's work assignments. Task or duty area qualification will be granted by line management based on successful evaluation of the worker's mastery of the learning objectives presented during training. Maintenance of qualification will be contingent upon successful completion of continuing training and/or through satisfactory training OJT evaluations. Criteria for exempting an employee from segments of classroom training or OJT are defined. Training materials are linked to the CM system to assure that design changes and modifications are accounted for in training. Programmatic and individual training and qualification records will be maintained in accordance with RMDC guidelines.

Plant functional organization managers develop and maintain a description of each individual's training requirements within their organization. These requirements will be identified in individual Training Requirement Matrices approved by supervision line and training management. Plant personnel, contractors, and visitors receive training as applicable to their positions or job function. The applicant describes the various training areas in Sections 11.3.1.1 - 11.3.1.12 of the LA (USEC, 2006b).

The staff finds that the applicant has provided an adequate description of the organization and management of the training function. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(1) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.2 Analysis and Identification of Functional Areas Requiring Training

In Section 11.3.2, "Analysis and Identification of Functional Areas Requiring Training," of the LA

(USEC, 2006b), the applicant states that a needs/job analysis will be used to identify the tasks affecting worker or public safety, safeguards of regulated material, or protection of the environment as identified in the ISA Summary (USEC, 2006a). The analysis will be conducted with subject matter experts and training personnel. The training programs for personnel in specified important plant job positions and worker classifications will be based on a needs/job analysis. A plant-specific task list will be developed for each of the specified important plant job positions. The tasks selected for training will be matrixed to the associated procedures and training materials. Procedure changes, equipment changes, job scope changes, facility modifications, and other changes affecting task performance will be monitored and evaluated for their impact on the development or modification of initial and continuing training programs. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.3 Position Training Requirements

Section 11.3.3, "Position Training Requirements," of the LA (USEC, 2006b) states that plant procedures and individual Training Requirement Matrices will delineate initial and continuing training requirements. The training program requirements for positions relied on for safety or personnel who perform actions that prevent or mitigate accident sequences described in the ISA Summary (USEC, 2006a) will be defined in Training Development and Administrative Guides (TDAGs). The TDAGs will include: (a) organizational and administrative responsibilities; (b) trainee selection criteria, including minimum educational, technical, experience, and physical requirements; (c) course loading for initial and continuing training; (d) test and evaluation guidelines; (e) training and evaluation documentation guidelines; and (f) training courses or modules for specific qualification areas. Positions and functions covered under the TDAGs, including minimum educational, technical, and experience, are identified in Sections 11.3.1.1 - 11.3.1.12 of the LA (USEC, 2006b). The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(3) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.4 Development of the Basis for Training, Including Objectives

Section 11.3.4, "Development of the Basis for Training, Including Objectives," of the LA (USEC, 2006b) states that learning objectives will be established to identify the training content and to define satisfactory trainee performance. Learning objectives state the requisite knowledge, skills, and abilities the trainee must demonstrate. The conditions under which the required actions take place and the standards of performance will be determined in developing the learning objectives. Learning objectives will be documented in lesson plans and training guides, and will be revised as necessary based on changes in procedures, plant systems/equipment, or job scope. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(4) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.5 Organization of Instruction, Using Lesson Plans and Other Training Guides

Section 11.3.5, "Organization of Instruction, Using Lesson Plans and Other Training Guides," of the LA (USEC, 2006b) states that learning objectives will be derived from related task lists and

analyzed. Classroom lesson plans, OJT guides, or other instructional materials will be procured or developed based on the instructional analysis. Lesson plans and other training guides will provide the guidance and training structure to ensure consistent delivery of training material. The lesson plans or other training guides will provide the evaluation tools to ensure mastery of the learning objectives. Classroom lesson plans, OJT guides, and other instructional materials receive technical reviews by designated subject matter experts and instructional reviews by training management as part of the approval process. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(5) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.6 Evaluation of Trainee Accomplishment

Section 11.3.6, “Evaluation of Trainee Learning,” of the LA (USEC, 2006b) states that within the job position/worker classification, training programs will provide logical instructional blocks or “modules” which will be presented in such a manner that specific learning objectives will be accomplished. Trainee progress will be evaluated by line and training management through a variety of performance demonstrations such as written examinations, oral examinations, and practical tests. Comprehensive qualification programs will contain periodic evaluations of trainee performance. Remediation will be provided as appropriate. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(6) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.7 Conduct of On-the-Job Training

Section 11.3.7, “Conduct of On-the-Job Training,” of the LA (USEC, 2006b) states that OJT will be conducted wherever practical in the work environment to demonstrate actual task performance. When the actual task cannot be performed, the task may be simulated. Applicable tasks and related procedures for each technical area provide OJT input that will be designed to supplement formal classroom training and ensure personnel will be qualified to perform their assigned tasks. Training requirements for specific job positions are discussed in Section 11.3.3 of the LA (USEC, 2006b). The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(7) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.8 Evaluation of Training Effectiveness

Section 11.3.8, ”Evaluation of Training Effectiveness,” of the LA (USEC, 2006b) states that systematic evaluations will be performed of training effectiveness and its relationship to on-the-job performance in order to ensure that the training program conveys the required knowledge and skills, and to revise the training, where necessary, based on the performance of trained personnel in the job setting. Plant design changes, modifications, or changes in task performance will be analyzed by line and training personnel for impact on training. Corrective actions and lessons learned will be factored into training. Student feedback of training received and line management evaluation of student performance on the job after training is complete will be used to determine training effectiveness. Post training evaluations will be requested of students and supervisors after completion of training. Line and training management will conduct self-assessments and evaluations of the individual training programs. QA auditors will provide additional assessments through the audit program. The assessments and evaluations will be used to determine training program strengths and weaknesses for continuous

improvement. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(8) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.9 Personnel Qualification

Section 11.3.9, "Personnel Qualification," of the LA (USEC, 2006b) states that the minimum education, experience, and qualification requirements for managers, engineers, technical professional staff, supervisors, technicians, and maintenance personnel will be identified and will be commensurate with the assigned functional responsibility and authority of the positions. Managers will have a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience. Engineers and other technical professional staff who affect the design, modification, operation, or maintenance of the IROFS identified in the ISA Summary (USEC, 2006a) will have a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and three years of nuclear experience. Supervisors of technicians, maintenance personnel, and other staff whose actions are relied on for safety will have a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and three years of industrial/chemical/nuclear plant operations, maintenance, engineering, or support experience. Plant maintenance personnel and technicians will have an associates degree in engineering or the physical sciences or equivalent technical experience, and three years of industrial/chemical/nuclear plant operations, maintenance, engineering, or support experience. Construction personnel, plant technicians, maintenance personnel, and other staff whose actions are relied on for safety will complete the applicable training programs or have equivalent experience or training. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(9) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.3.10 Provisions for Continuing Assurance

Section 11.3.10, "Provisions for Continuing Assurance," of the LA (USEC, 2006b) states that continued training and periodic re-qualification will be provided based on the frequency required by regulatory agencies and national standards, overtrain tasks identified in performance-based training programs, and additional training as determined by line management that includes, but will be not limited to: (a) nuclear criticality safety assessments; (b) plant or system changes; (c) component changes; (d) procedure changes; (e) lessons learned; and (f) emergency response procedures. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.3(10) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.4 Procedure Development and Implementation

Section 11.4, "Procedures," of the LA (USEC, 2006b) states that activities involving nuclear material and/or IROFS will be conducted in accordance with written procedures. Procedures related to the operation of IROFS where human actions are important and for the management measures supporting those IROFS are governed by the requirements of this section. The two general types of procedures are Operating and Administrative. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(1) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

Operating procedure elements will address: (1) the purpose of the activity; (2) regulations, policies, and guidelines governing the procedure; (3) type of procedure, steps for each

operating process phase; (4) initial start-up; (5) normal operations; (6) temporary operations; (7) emergency shutdown; (8) emergency operations; (9) normal shutdown; (10) startup following an emergency or extended downtime; (11) hazards and safety considerations; (12) operating limits; (13) precautions; (14) measures to be taken if contact or exposure occurs; (15) IROFS associated with the process and their functions; and (16) the time frame for which the procedure will be valid. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

Maintenance procedures for corrective and preventive maintenance, functional testing after maintenance, and surveillance maintenance activities will describe: (1) the qualifications of personnel authorized to perform maintenance or surveillance; (2) controls on and specification of replacement components or materials to be used; (3) post-maintenance testing to verify operability of the equipment; (4) tracking and records management of maintenance activities; (5) safe work practices; (6) pre-maintenance activities, including procedure reviews for accuracy and completeness; and (7) steps for notification of affected parties before performing work and upon completion of work, including discussion of potential degradation of IROFS during the planned maintenance. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(11) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

Administrative procedures or "management control" procedures will be used to manage activities involving: (1) design; (2) configuration management; (3) procurement; (4) construction; (5) radiation safety; (6) maintenance; (7) QA elements; (8) training and qualification; (9) audits and assessments; (10) incident investigations; (11) records management; (12) criticality safety; (13) fire safety; (14) chemical process safety; and (15) reporting requirements. The information provided by the applicant, as described above, meets the guidance in Sections 11.4.3.4(3) and (5) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The applicant will develop or modify procedures through a formal process incorporating the change controls described in Section 11.1 of the LA (USEC, 2006b). The procedural modification process will ensure that: (1) procedures will be identified and developed as needed, including for those operations of IROFS where human actions are necessary and for supporting management measures; (2) the procedures will be approved under the guidelines of the CM program by personnel responsible and accountable for the operation; (3) the procedures will be verified and validated through field tests to provide assurance that they are usable and accurate, and will be periodically reviewed and re-verified and validated; (4) operating limits, safety limits and IROFS will be clearly identified; (5) required actions for off-normal conditions as well as normal operations will be included; (6) hold points or safety checkpoints will be identified at appropriate steps in the procedure; (7) a mechanism will be specified for revising and reissuing procedures in a controlled manner; (8) current procedures will be available and used at work locations; and (9) the Plant Training Program will train the required persons in the use of the latest available procedures. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(4) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

Section 11.4.9, "Topics to be Covered in Procedures," of the LA (USEC, 2006b) identifies the site-wide safe work practices to control processes and operations with licensed material, IROFS, and hazardous chemicals in topics that will be covered by procedures pertaining to the general areas of administrative procedures, system procedures that address start-up, operation

and shutdown, abnormal operation/alarm response, maintenance activities that address system repair, calibration, inspection and testing, and emergency response procedures. The procedure hierarchy described in Section 11.4.3, "Procedure Hierarchy," of the LA (USEC, 2006b) includes four levels for policy statement, standard practice procedures applicable to one organization, procedures issued at the organizational level, and procedures issued within a group. Site-wide safe work practices will be included in maintenance procedures. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(6) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The applicant states that procedures will be reviewed following unusual incidents, such as an accident, unexpected transient, significant operator error, or equipment malfunction, to determine if changes are appropriate based on the discovery of root cause and corrective action determination for the particular incident. Procedure changes that are necessary because of a system modification will be addressed as part of the modification control process. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(7) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The applicant will conduct a verification process to ensure that procedures are technically adequate and can be performed as written. The process will ensure that technical information including formulas, set points, and acceptance criteria are correctly identified. The procedure will undergo a walk-down in the field or a tabletop walk-through. Drafts of new procedures and procedure changes will receive technical reviews, safety reviews, and cross-discipline reviews, as needed. During the development process, input and review by affected parties will be required, including QA to ensure that QA requirements are identified and included in operating procedures. The applicant states that the formal procedure process will ensure that operating limits and IROFS will be specified in the procedures. The Plant Safety Review Committee will review each new procedure or change involving the operation of IROFS and the management measures supporting those IROFS, operator actions necessary to prevent or mitigate the consequences of accidents described in the ISA Summary (USEC, 2006a), and safe work practices involving special nuclear material, IROFS, and hazardous chemicals incident to the processing of licensed material. Approval authority rests with the applicable ACP organization manager responsible for the activity. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(8) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

The applicant states that a procedure validation process will be used to ensure that no technical errors or human factor issues were inadvertently introduced during the procedure review process. Validation will be required for new procedures and intent changes. Validation will be performed in the field by qualified personnel. If the particular system or process is not available for a walk-through validation, a talk-through may be performed in the particular shop or training environment. Performance of procedure validation will be documented.

Procedures will be issued and controlled in accordance with the RMDC program procedures, described in Section 11.7 of the LA (USEC, 2006b). The program will ensure that documents are controlled and distributed in accordance with written requirements and authorizations. Copies of current approved procedures will be available to users via electronic and/or hard copy distribution in the work area. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(9) of NUREG-1520 (NRC, 2002), and is therefore

acceptable.

Temporary procedures may be issued when permanent procedures do not exist to direct operations during testing, maintenance, and modifications which will provide guidance in unusual situations not within the scope of permanent procedures, and to ensure orderly and uniform operations for short periods when an item's performance is not covered by existing permanent procedures, or when portions of existing procedures do not apply. Temporary procedures can be used for a short period of time, which should not exceed 60 days without an additional assessment of the procedure. These procedures will receive the same level of review and approval as required for permanent procedures.

Temporary changes to procedures involving the operation of IROFS and the management measures supporting those IROFS, operator actions necessary to prevent or mitigate the consequences of accidents described in the ISA Summary (USEC, 2006a), or safe work practices involving special nuclear material, IROFS, or hazardous chemicals incident to the processing of licensed material, may be made, provided: (1) the change does not result in a change to the ISA as documented by a 10 CFR 70.72 review; (2) the temporary change does not constitute an intent change (i.e., a change in scope, method, or acceptance criteria used that has safety significance); and (3) the change is documented. Temporary changes to procedures should not exceed 30 days. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(10) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

Emergency operating procedures, alarm response procedures, and procedures dealing with highly hazardous chemicals as defined by the chemical safety program will be reviewed on a 1-year cycle. Procedures not included in the 1-year cycle will be reviewed every 5 years. The procedure owner or subject matter expert will perform a complete administrative and technical review of the procedure to ensure that the information contained within the procedure will be complete and accurate and the procedure will be usable as written. In-hand procedures (continuous use) will be followed step-by-step and will be present in the work area while the tasks are being performed. Reference use procedures cover routine procedural actions that will be frequently repeated or of minimal complexity, and can be performed from memory. If a step in the procedure cannot be performed as written, work will be stopped, the system or process immediately placed in a safe condition, and corrective actions initiated. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.4(12) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.5 Audits and Assessments

11.3.5.1 Audits and Assessments Policy

Section 11.5, "Audits and Assessments," of the LA (USEC, 2006b) describes the implementation of a system of audits and assessments used to ensure that the health, safety, and environmental programs, as described in the LA (USEC, 2006b) will be adequately and effectively implemented. The system will be designed to ensure comprehensive program oversight at least every three years for all ACP activities and functions. The system will be comprised of audits and assessments.

11.3.5.2 Audits

Section 11.5.1, “Audits,” of the LA (USEC, 2006b) states that qualified auditors in the QA organization will conduct audits in accordance with written procedures or checklists. The auditing organizations will be independent from operations of the plant. Audits will verify the effectiveness of health, safety, and environmental programs and their implementation, and determine the effectiveness of the process being assessed. Audits will further verify that plant operations will be conducted safely and in accordance with regulatory requirements and LA (USEC, 2006b) commitments.

Audits will be conducted in accordance with Section 18 of the QAPD (USEC, 2006c), and will use written procedures or checklists. Audits will be performed under the direction of a Lead Auditor, qualified in accordance with the American National Standards Institute (ANSI)/American Society of Mechanical Engineering (ASME) NQA-1, Supplement 2S-3 (ASME, 1984). Lead Auditors and staff auditors will be functionally and organizationally independent of the programs and activities that will be examined.

In addition to periodically evaluating aspects of the QAPD (USEC, 2006c), the applicant states that audits will be conducted in the areas of: (a) radiation safety; (b) nuclear criticality safety; (c) chemical safety; (d) fire safety; (e) environmental protection; (f) emergency management; (g) QA; (h) CM; (i) maintenance; (j) training and qualification; (k) procedures; (l) incident investigation; and (m) records management.

The applicant will document and report audit results to plant senior management as specified in facility procedures. Provisions will be made for reporting and corrective action, where warranted. The plant Corrective Action Program, described in Section 11.6, “Incident Investigation,” of the LA (USEC, 2006b), will be administered by the Regulatory Organization to ensure proper control of corrective actions as defined in Section 16 of the QAPD (USEC, 2006c).

11.3.5.3 Assessments

Section 11.5.2, “ Assessments,” of the LA (USEC, 2006b) states that management responsible for implementing portions of the QAPD (USEC, 2006c) will be required to perform assessments to verify the adequacy of the part of the QAPD (USEC, 2006c) for which they will be responsible and to ensure its effective implementation. Personnel from the area being assessed may perform the assessment, provided that they do not have direct responsibility for the specific activity being assessed. Results of assessments will be documented, and the findings and observations will be resolved.

Organization managers maintain an assessment process within their organization to assess the adequacy and effectiveness of the implementation of the programs under their cognizance. As a minimum, these assessments will be conducted for the areas of radiation safety, nuclear criticality safety, chemical safety, fire safety, environmental protection, emergency management, QA, CM, maintenance, training and qualification, procedures, incident investigation, and records management.

Assessment results will be documented and reported as specified in plant procedures. Provisions will be made for reporting and corrective action, where warranted, in accordance with the ACP Corrective Action program.

11.3.5.4 Audits and Assessments Conclusion

The staff finds that the applicant has provided reasonable assurance that audits and assessments will be adequately performed and documented. The information provided by the applicant, as described above, meets the guidance in Sections 11.4.3.5(1) through (6) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.6 Incident Investigations

Section 11.6, "Incident Investigations," of the LA (USEC, 2006b) describes the incident investigation process for the identification, reporting, and investigation of abnormal events. The process incorporates the reporting requirements of 10 CFR 70.50 and 70.74. The applicant will establish procedures to assure that abnormal events and conditions will be promptly reported to appropriate personnel, assessed, and, when required, reported to NRC. The process will involve reporting to line management or directly to the Operations Supervisor. The Operations Supervisor will assess and categorize the event by procedure using the notification and reporting criteria of 10 CFR 70.50 and 70.74.

The level of investigation will be based on a graded approach relative to the severity of the event and will be conducted and documented in accordance with procedures. Each reportable event where a follow-up written report to NRC is required will be investigated to determine the root cause and corrective actions necessary to prevent reoccurrence. Other events not requiring a written report will be evaluated using the Corrective Action program to determine the appropriate actions.

A prompt, risk-based evaluation will be performed and an investigation will be initiated within 48-hours of the event, or sooner, depending on the safety significance. A procedure will provide a documented plan for conducting the investigation. The plan is separate from the Emergency Plan. A reasonable, systematic, structured approach will be used to determine the specific or generic root causes of the event. A record of IROFS failures required to be maintained by 10 CFR 70.62(a)(3) will be reviewed as part of the investigation. Auditable records and documentation related to the investigation will be kept for at least two years or the life of the operation, which ever will be longer. The original investigation reports will be available to NRC upon request.

The investigators will be independent of the line function involved in the incident under investigation and will have the authority to obtain all of the information necessary in order to conduct the investigation. Line management will fully cooperate with the investigators. The individual leading the investigation will be trained and qualified in root cause analysis techniques. If a team is to be used, at least one member will be a process expert.

Records of IROFS failures will be kept and updated in accordance with 10 CFR 70.62(a)(3). Record revisions necessitated by the post-failure investigation and conclusions, will be made promptly in accordance with 10 CFR 70.62(a)(3).

Corrective actions will be developed, tracked in a database, and monitored through completion for significant conditions adverse to quality or when a follow-up written report to NRC is required. Corrective actions will be taken within a reasonable time, commensurate with safety

significance of the event. Record revisions will be made promptly and proof to support action closure will be maintained. Details of the event will be compared with the accident sequences already considered by the ISA. When necessary, the ISA Summary (USEC, 2006a) will be updated and relevant findings will be reviewed by ACP personnel.

The staff finds that the applicant has provided an adequate description of its incident investigation process. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.6 of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.7 Records Management

Section 11.7.3.1, "Records Management and Document Control," of the LA (USEC, 2006b) states that the Engineering Manager will be responsible for the RMDC program. Designated responsibilities include: (a) directing the activities and personnel of the RMDC programs; (b) directing the development; (c) implementation; and (d) maintenance of methods and procedures encompassing a records management program and a document control program, and assuring that the laws, codes, standards, regulations and company procedures pertaining to record keeping and document control requirements will be met. Administrative controls for the generation and revision of records will be contained in implementing procedures. The Records Management program will assure that appropriate records of IROFS will be maintained in accordance with the BDC of 10 CFR 70.64(a) and the defense-in-depth requirements of 10 CFR 70.64(b). Reproduced records will be authenticated by properly authorized personnel. Records will also be categorized and handled with respect to their relative importance to safety and storage needs.

Examples of records maintained by the RMDC are listed in Section 11.7.5 of the LA (USEC, 2006b). The requirements for records management will vary according to the nature of the plant and the hazards and risks posed by it. The listed examples cover the records required by 10 CFR Parts 19, 20, 21, 25, and 70, but are not intended to be exhaustive or prescriptive, as different or additional records may be required in certain circumstances.

Record retention time will be specified in a schedule developed by the manager of the organization that originates the record, or their designee. The process for disposition of records at the end of their retention lifetime will be specified in procedures. Documents designated to become records will be required to be legible, accurate, complete, and contain an appropriate level of detail commensurate with the work being performed and the information required for that type of record.

Records will be protected from deterioration, loss, damage, theft, tampering, and/or unauthorized access for the life of the record. Requirements for controlling access and maintaining accountability for records will be provided, to assure that only authorized personnel have access to those records to prevent loss, damage, or inadvertent destruction. Instructions for the protection of special record media also will be provided to prevent damage from conditions adverse to their preservation.

Records will be authenticated and validated by the manager of the organization that originates the record, or their designee, as specified in the procedure that controls the generation and revision of these records. Instructions will be provided to the record originator for record protection until they are formally transmitted to Records Management. Records designated in

the QAPD (USEC, 2006c) will be stored in authorized facilities or containers to provide protection from hazardous conditions. Requirements will cover both permanent and temporary storage.

Controlled documents will be distributed in accordance with controlled distribution lists to assure that they will be available at locations where work will be performed. Specific time requirements will be established for controlled document distribution and receipt acknowledgment. Document Control will track the distribution and receipt of transmittals. A controlled document index will be created and maintained by Document Control, for each document, or document type, and traceable to the latest approved revision level. The generators of controlled documents will be required to notify Document Control whenever a document will be voided, canceled, or superseded. Document Control will remove the document from distribution and notify copyholders of the document's changed status.

Computer codes used in the Document Control program will be controlled and maintained in accordance with the "Computing and Telecommunications Security Manual" and Information Systems procedures. Virus protection and access control to the document control program database will be provided to ensure continuing usability of the codes and data as technology changes. Procedures will allow older forms of information and codes from older computing equipment to be transferred. Routine Document Control database backups will be performed.

Records of IROFS failures will be kept and updated in accordance with 10 CFR 70.62(a)(3). Record revisions necessitated by post-failure investigation conclusions will be made promptly in accordance with 10 CFR 70.62(a)(3).

The applicant stated that the overall effectiveness of the Records Management program will be evaluated through the audit program established in Section 18 of the QAPD (USEC, 2006c). Identified deficiencies will be corrected in a timely manner in accordance with the procedures described in Section 11.6 of the LA (USEC, 2006b).

The staff finds that the applicant has provided an adequate description of its record management and document control program. The information provided by the applicant, as described above, meets the guidance in Sections 11.4.3.7(1) through (5) and Appendix B of Chapter 11 of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8 Other QA Elements

Section 11.8, "Other QA Elements," of the LA (USEC, 2006b) states that the applicant has developed QA principles that will apply to the design, fabrication, refurbishment, modification, testing, operation, and maintenance of the ACP in order to meet the requirements of 10 CFR 70.64(a)(1). These principles are described in the applicant's QAPD.

11.3.8.1 Organization

Section 1.0, "Introduction," of the QAPD (USEC, 2006c) describes: (a) the organizational structure; (b) functional responsibilities; and (c) charts of the lines, interrelationships, and areas of responsibility and authority for all organizations performing activities relied on for safety. The Vice President, American Centrifuge, has overall responsibility for the design, fabrication,

refurbishment, testing, operation, and modification of the ACP, and will also be also responsible for the QA program and for determining the status, adequacy, and effectiveness of the QAPD (USEC, 2006c). The Director, American Centrifuge Plant reports to the Vice President, American Centrifuge, and will be responsible for the overall safe operation and maintenance of the ACP. These responsibilities include refurbishment/construction, initial start-up, testing operation, training, procedures, engineering, as well as occupational, environmental, and nuclear safety. The Director, Regulatory and Quality Assurance, reports to the Vice President, American Centrifuge, and will be responsible for regulatory and QA functions. The QA Manager reports to the Director, Regulatory and Quality Assurance, and will be responsible for independent oversight of ACP activities covered by the QAPD (USEC, 2006c). Organizational responsibilities for design, refurbishment, construction, start-up, and operation are functionally described as follows:

The Engineering Manager reports to the Director, American Centrifuge Plant. The Engineering Manager is responsible for site characterization; plant design and the design control process; configuration management; engineering; and acceptance test coordination, including test control. The Engineering Manager is also responsible for nuclear criticality safety, safety analysis, records management and document control, and approving disposition of nonconforming items when dispositioned as “repair” or “use-as-is.”

The Production Support Manager reports to the Manager, Enrichment Operations. The Production Support Manager is responsible for the Radiation Protection Program; industrial safety; industrial hygiene; chemical safety; waste management; environmental survey; and implementing the training and procedures programs.

The Operations Manager reports to the Manager, Enrichment Operations. The Operations Manager is responsible for enrichment operations; feed and withdrawal operations; utilities; production management; shift operations; packaging and transportation; and repair and assembly of centrifuge machines.

The Maintenance Manager reports to the Manager, Enrichment Operations. The Maintenance Manager is responsible for safe and reliable performance of preventive and corrective maintenance and support services on buildings/facilities and equipment, with the exception of centrifuge machines, and for integrated planning and scheduling.

The Director, Regulatory and Quality Assurance, reports to the Vice President, American Centrifuge. The Director, Regulatory and Quality Assurance, is responsible for the management of the regulatory and QA functions and the ACP policy system. This individual is the primary day-to-day interface with NRC and has overall responsibility for management of activities related to license requirements for the ACP.

The Regulatory Manager reports to the Director, Regulatory and Quality Assurance. The Regulatory Manager is responsible for regulatory oversight functions, environmental compliance, plant change process, commitment management, and the Corrective Action program.

The Plant Support Manager reports to the Director, American Centrifuge Plant. The Plant Support Manager is responsible for fire safety, health services, emergency management, and nuclear materials control and accountability for the ACP.

The Procurement Manager is responsible for procurement; providing procurement material control services (including supplier qualification coordination, purchasing, contracting, receiving and control of nonconforming items); and material control (including handling, storage and shipping). This manager is also responsible for supply strategy and development of qualified long-lead-time and complex-system suppliers.

Persons and organizations responsible for ensuring that appropriate QA will be established for verifying that activities affecting quality have been correctly performed will have sufficient organizational freedom to identify problems, initiate solutions, and will verify solutions and control further processing when necessary.

The staff finds that the applicant has provided an adequate description of the QA organization element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(1) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.2 QA Program

Section 2.0, "Quality Assurance Program," of the QAPD (USEC, 2006c) describes the QA elements that will be applied to the design, fabrication, testing, operation, procurement, inspection, maintenance, and modifications of IROFS and activities affecting those IROFS to assure that they will be available and reliable to perform their safety function when needed. The QAPD (USEC, 2006c) will be applied to IROFS in a graded manner, commensurate with their importance to safety. QLs will be established as follows:

QL-1 A single IROFS that prevents or mitigates a high consequence event.

QL-2 Two or more IROFS that prevent or mitigate a high consequence event; or one or more IROFS that prevents or mitigates an intermediate consequence event.

QL-3 Any item other than QL-1 and QL-2. QL-3 items will be controlled in accordance with standard commercial practices.

The requirements of the QAPD (USEC, 2006c) will be applied in total to QL-1 IROFS. For QL-2 IROFS, implementing procedures will provide for graded application of QA elements taking into consideration: (a) QL (risk significance); (b) applicable regulations, industry codes and standards; (c) complexity or uniqueness of an item or activity and the environment in which it has to function; (d) quality history of the item in service; (e) degree to which functional compliance can be demonstrated or assessed by test, inspection, or maintenance methods; anticipated life span; (f) degree of standardization; (g) importance of data generated; (h) reproducibility of results; and (i) the consequences of failure. QL-3 items are not covered by the applicant's QAPD. Application of the QAPD will be documented, planned, implemented and maintained to provide reasonable assurance that, together with other management measures, IROFS will be available and reliable when needed.

The results of the application of the graded approach to quality will be incorporated into design requirement documents, specifications, procedures, instructions, drawings, inspection plans, test plans, procurement documents, and other documents that establish the requirements for items and activities.

Indoctrination and training of personnel performing or managing activities affecting quality will meet the requirements of Part I of ANSI/ASME NQA-1 Supplement 2S-4, “Supplementary Requirements for Personnel Indoctrination and Training” (ANSI/ASME, 1994). Quality Control personnel performing inspection and testing will meet the requirements of Part I of ANSI/ASME NQA-1 Supplement 2S-1, “Supplementary Requirements for the Qualification of Inspection and Test Personnel” (ANSI/ASME, 1994). Personnel performing nondestructive examination will meet the requirements of SNT-TC-1A, “American Society for Nondestructive Testing Recommended Practice” (ASNT, 1980). QA audit personnel will meet the requirements of ANSI/ASME NQA-1, Part 1, Supplement 2S-3, “Supplemental Requirements for the Qualification of Quality Assurance Program Audit Personnel” (ANSI/ASME, 1994).

Regular assessments of the adequacy of the QAPD (USEC, 2006c) and effective implementation of the QA elements will be performed by the management of the organizations implementing the QAPD (USEC, 2006c) and by the responsible senior managers.

The staff finds that the applicant has provided an adequate description of the QA program element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(2) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.3 Design Control

Section 3.0, “Design Control,” of the QAPD (USEC, 2006c) describes the design control process and procedures that include design inputs, process, analyses, verification, interfaces, changes, and design documentation and records.

Design Inputs

Design inputs are derived from design process requirements and approved procedures provide instructions for performing the design process in a planned, controlled, and systematic manner. The ACP design control process includes performance standards and technical content derived from the ISA and Management Measures.

Design inputs, such as design bases, performance requirements, regulatory requirements, codes and standards, are identified and documented as design requirements (e.g., primary requirements, functional requirements, and system requirements). These design based documents are reviewed and approved in a timely manner and to the level of detail and consistency necessary to permit the design activity to be carried out correctly and to provide a consistent basis for making sound engineering and technical design decisions, which are translated through design verification measures, and evaluation of design changes and facility configuration. Changes, including the reason for the changes and whether or not prior NRC approval is required to make the changes, are identified, approved, documented, and controlled.

Design Control Process

The design control process is implemented through the use of approved procedures which provide consistency in the performance of design activities in a planned, controlled, and documented manner. The design control process will also be applied to the ISA and management measures. Design inputs, such as design bases, performance requirements,

regulatory requirements, codes, and standards, are identified and documented as design requirements. These design methods, which are comprised of materials, parts, equipment, and processes that are essential to determining the form, fit, and function of the IROFS are selected and reviewed for suitability of application. Final design output documents, including changes, are correlated to the design input to permit design verification. Computer Program design outputs are developed, validated, and managed in accordance with the applicant's commitment to Basic Requirement 11, "Test Control," of ANSI/ASME NQA-1 and Part II, Subpart 2.7, "QA Requirements for Computer Software for Nuclear Facility Applications" (ANSI/ASME, 1994). The applicant states that internal and external design interfaces will be identified and controlled, and design efforts will be coordinated among participating organizations. Design information transmitted across interfaces will be reviewed, approved, documented, and controlled. Final design documentation and records that provide evidence that the design and design verification processes were performed in accordance with the QAPD (USEC, 2006c) will be collected, stored, and maintained.

Design process activities are planned on a timely basis and to the level of detail necessary to permit the design process to be carried out correctly; to permit verification that the design inputs are correctly translated into design documents; and to support interfacing design, procurement, fabrication, and operation. Appropriate quality standards are identified and documented. Changes from specified quality standards, including the reasons for the changes and whether or not prior NRC approval is required to make the changes, are identified, approved, documented, and controlled. Design methods, materials, parts, equipment, and processes that are essential to the function of the IROFS are selected and reviewed for suitability of application. Assemblies, subassemblies, and parts are clearly identified. Commercial grade items that have been modified or which need to meet special verification requirements are uniquely identified.

Final design output documents, including changes thereto, are relatable to the design input by documentation in sufficient detail to permit design verification.

Design Analysis

Design analyses documents (e.g., calculations) contain sufficient detail as to the purpose, method, assumptions, design input, references, and units such that a person technically qualified in the subject can understand the analyses and verify the adequacy of the results without recourse to the originator. Design analysis, performed with computer systems, will list the software and version; hardware; inputs and outputs; and evidence of computer program verification/validation or alternate verification of the results. Design analysis documents are identifiable by subject, originator, reviewer, and date or by other identification such that the documents are retrievable.

Design Verification

Design verification is performed and documented, in accordance with approved procedures, by competent individuals or groups other than those who performed the original design. The extent and method of the design verification is a function of the importance to safety, the complexity of the design, the degree of standardization, the state of the art, past performance, and similarity with previous proven designs. Where changes to previously verified designs are made, design

verification is performed for the changes, including an evaluation of the effects of the changes on the overall design and on any design analysis on which the design is based. Methods of design verification include any one or a combination of the following (as defined in Supplement 3S-1 of ANSI/ASME NQA-1): design reviews, alternate calculations, or the performance of qualification tests. Verification by testing is performed when deemed necessary and demonstrates adequacy of performance under conditions that simulate the most adverse design requirements. Verification of computer programs includes appropriate testing and validation. Design verification is performed in a timely manner and is completed before relying upon the IROFS, or computer program to perform its function.

Verifiers are knowledgeable in the areas to be verified. The verifier may be a supervisor, provided the supervisor was not directly responsible for the design (i.e., did not specify a singular design approach or rule out certain design consideration and did not establish the design inputs used in the design) or provided the supervisor is the only individual in the organization competent to perform the verification. However, verification is more than a cursory supervisory review. A supervisor with direct responsibility for the design may verify QL-2 items and services.

Design Change Control

Changes to final designs, field changes, modifications, and non-conforming items dispositioned "use-as-is" or "repair" are justified, documented, and subject to the design control measures commensurate with the original design. Changes are reviewed and approved by the person or group with assigned design authority. Changes to designs that have been approved or certified by NRC (e.g., 10 CFR Part 71 package design) are subject to the necessary additional controls.

Design Interface

Internal and external design interfaces are identified and controlled and design efforts are coordinated among participating organizations. Design information transmitted across interfaces is reviewed, approved, documented, and controlled.

Design Documentation and Records

Final design documentation and records that provide evidence that the design and design verification processes were performed in accordance with this section are collected, stored, and maintained.

The staff finds that the applicant has provided an adequate description of the design control element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(3) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.4 Procurement Document Control

Section 4.0, "Procurement Document Control," and Section 7.0, "Control of Purchased Items and Services," of the QAPD (USEC, 2006c) describe the control of the procurement process, procurement documents, and procured material, components, and services. The applicant states that procurement documents will include those requirements necessary to assure that items and services to be provided will be of desired quality. The documents will include, as

appropriate: (a) the scope of work; basic technical requirements; (b) QA requirements, including the requirements for the supplier to have an acceptable QA program consistent with the applicable portions of the QAPD; (c) requirements for control of non-conformances and changes; (d) requirements for subtier suppliers; and (e) documentation requirements. Procurement documents and changes thereto will be reviewed to ensure that they contain appropriate requirements. Changes to procurement documents will be subject to the same control as the original.

The staff finds that the applicant has provided an adequate description of the procurement document control element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(4) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.5 Instructions, Procedures, and Drawings

Section 5.0, "Instructions, Procedures, and Drawings," of the QAPD (USEC, 2006c) states that activities affecting the availability and/or reliability of IROFS will be prescribed by and performed in accordance with documented instructions, procedures, and drawings of a type appropriate for the circumstances. These documents include or reference appropriate acceptance criteria for determining if prescribed activities have been satisfactorily accomplished. The applicant's QA organization will review QA implementing procedures for compliance and consistency with the QAPD (USEC, 2006c) and ensures that the QAPD (USEC, 2006c) provisions will be effectively incorporated into QA implementing procedures.

The staff finds that the applicant has provided an adequate description of the instructions, procedures, and drawings element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(5) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.6 Document Control

Section 6.0, "Document Control," of the QAPD (USEC, 2006c) describes the document control QA element. The preparation, issuance, and modification of documents that specify quality requirements or prescribe activities affecting the availability and/or reliability of IROFS will be controlled to provide assurance that the appropriate documents will be in use. Document changes will be reviewed for adequacy and approved for implementation by authorized personnel. Procedures and instructions assure that documents will be prepared, reviewed for adequacy, correctness, and completeness by a qualified individual, approved for release by authorized personnel, distributed to the location where the activity will be performed before commencing work, and used in performing the activity. The procedures will require the creation and maintenance of a controlled document index to track and control approved document revision levels. Unless other organizations will be specifically designated, changes to documents will be reviewed and approved in the same manner as the original.

The staff finds that the applicant has provided an adequate description of the document control element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(6) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.7 Control of Purchased IROFS and Services

Section 7.0, “Control of Purchased Items and Services,” of the QAPD (USEC, 2006c) states that the procurement of items and services will be controlled to assure conformance with specified requirements. Controls will provide for, as appropriate: source evaluation and selection; evaluation of objective evidence of quality furnished by the supplier; source inspection; audit; and examination of items or services upon delivery or completion. Requirements will cover the dedication of commercial grade items or services.

Supplier selection for noncommercial grade items and services will be based on an evaluation that considers, as applicable: (a) the potential supplier’s history; (b) the supplier’s QA program; (c) the results of recognized industry shared supplier audits; (d) valid National Voluntary Laboratory Accreditation Program Certificates of Accreditation; (e) whether the supplier maintains and implements an NRC approved QA program; and (f) whether the supplier maintains a valid ASME Code certification. Measures will be established to interface with the supplier and verify the supplier’s performance, as necessary. Acceptance of items will include one or more of the following: (1) Certificate of Conformance; (2) source verification; (3) receiving inspections; (4) post-installation testing; and (5) supplier qualification and performance history.

Acceptance of suppliers of commodities and services are based on acceptable technical, quality and commercial qualifications. Acceptable suppliers are placed on the Approved Suppliers List (ASL) maintained by the QA organization. Retention on the ASL is based on maintaining acceptable performance. Suppliers that are not pre-qualified may be used with appropriate compensatory controls as agreed upon by the QA organization.

Commercial grade items and services will be subject to design control measures. Criteria and methods for identifying the characteristics for acceptance will be established to provide reasonable assurance that the item or service provided meets specified requirements. Receipt inspections will be performed as a minimum for acceptance of commercial grade items. Commercial grade services will be subject to acceptance reviews. Dedication of a commercial grade item or service will be based on the above, as well as any additional requirements imposed by Engineering, based on the complexity of the item or service, or its importance to safety.

The staff finds that the applicant has provided an adequate description of the control of purchased IROFS and services element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(7) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.8 Identification and Control of IROFS

Section 8.0, “Identification and Control of Items,” of the QAPD (USEC, 2006c) describes the IROFS that will be identified and controlled, as necessary, from initial receipt and fabrication of the items up to and including installation and use to provide assurance that only correct and acceptable items will be used or installed. Physical identification will be used to the maximum extent possible. Traceability of items to specific records will be provided when specified by codes, standards, or specifications. Items with a limited shelf or operating life will be identified and controlled and procedures will be used to provide for item identification consistent with the

planned duration and conditions of storage.

The staff finds that the applicant has provided an adequate description of the identification and control of IROFS element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(8) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.9 Control of Special Processes

Section 9.0, "Control of Special Processes," of the QAPD (USEC, 2006c) describes the control of processes affecting quality of items and services. Special processes that control or verify quality (i.e., those used in welding, heat treating, and nondestructive examination) will be performed by qualified personnel using qualified procedures in accordance with specified requirements, codes, or standards. Certification of individuals, qualification of processes, and/or control of process parameters, equipment, calibration or acceptance criteria will be prescribed when necessary. Records will be maintained of currently qualified personnel, processes, and equipment for special processes.

The staff finds that the applicant has provided an adequate description of the control of special processes element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(9) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.10 Inspections

Section 10.0, "Inspection," of the QAPD (USEC, 2006c) describes the QA element of inspection. The applicant will plan and perform inspections in order to verify conformance of items to applicable requirements. Inspection requirements will be based on the importance of the item or activity, and will be specified in written procedures with provisions included for documenting and evaluating inspection results. Personnel qualification programs will be established for inspection personnel. Inspection records will contain information related to the item, inspector, type of observation or inspection plan, results of the inspection or acceptability of the item, and actions taken with regard to any non-conformances.

The staff finds that the applicant has provided an adequate description of the inspections element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(10) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.11 Tests

Section 11.0, "Test Control," of the QAPD (USEC, 2006c) describes the tests that will be performed in order to verify conformance to specified requirements, to demonstrate satisfactory performance, or to collect data. These tests will include design verification tests, acceptance tests, pre-operational tests, post-maintenance tests, and operational tests. Test requirements will be specified in written procedures with provisions included for documenting and evaluating test results. Test records will contain the item tested, test date, tester, type of observation, test procedure, results and acceptability, actions taken on any deviations, and person evaluating the

results. Personnel qualification programs will be established for test personnel. Computer software tests are developed, validated, verified, and managed in accordance with ASME NQA-1, 1994 edition, Basic Requirement 11 and NQA-1, Part II, Subpart 2.7, QA Requirements for Computer Software for Nuclear Plant Applications.

The staff finds that the applicant has provided an adequate description of the tests element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(11) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.12 Control of Measuring and Test Equipment

Section 12.0, "Control of Measuring and Test Equipment," of the QAPD (USEC, 2006c) describes the controls to provide reasonable assurance that tools, gauges, instruments, and other measuring and testing devices will be properly identified, controlled, calibrated, and adjusted at specified intervals to maintain performance within required limits. This section appropriately specifies requirements for calibration procedures, a control system for items requiring calibration, application and actual item use documentation, out-of-calibration actions, and records.

The staff finds that the applicant has provided an adequate description of the control of M&TE element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(12) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.13 Handling, Storage, and Shipping

Section 13.0, "Handling, Storage, and Shipping," of the QAPD (USEC, 2006c) describes the measures to handle, store, and ship material and equipment in accordance with design and procurement requirements to protect against damage, deterioration, or loss. Special coverings, equipment, and protective environments will be specified and provided where necessary for the protection of particular items from damage or deterioration. Special handling tools and equipment will be provided where necessary and controlled and maintained so that they will be ready and fit to serve the intended function. Operators of special equipment will be experienced or trained as necessary. Special handling, preservation, storage, cleaning, packaging, or shipping instructions will be established and used when essential to maintain acceptable quality.

The staff finds that the applicant has provided an adequate description of the handling, storage, and shipping element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(13) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.14 Inspection, Test, and Operating Status

Section 14.0,"Inspection, Test, and Operating Status," of the QAPD (USEC, 2006c) states that procedures will be established to ensure that the status of inspection and test activities will be either marked or labeled on the item or in documentation traceable to the item. This activity will be performed when it is necessary to ensure that required inspections and tests will be performed, and to ensure that items that have not passed the required inspections and tests will not be inadvertently installed, used, or operated. Status indicators will be used when required, and authority for the application and removal of status indicators will be specified.

The staff finds that the applicant has provided an adequate description of the inspection, test, and operating status element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(14) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.15 Control of Nonconforming IROFS

Section 15.0, "Control of Nonconforming Items," of the QAPD (USEC, 2006c) states that items and related activities that do not conform to specified requirements will be controlled to prevent inadvertent installation or use. Nonconforming items will be segregated and dispositioned as required. The responsibility and authority for evaluation and disposition of nonconforming items will be defined. The disposition process will include consideration of the need for design documents to be designated "as-built" to facilitate operations, maintenance, or modification. Repaired or re-worked items will be re-examined in accordance with the original acceptance criteria, unless an alternative acceptance criteria has been established. Documentation of the nonconformance includes its description, disposition, and disposition signatures.

The staff finds that the applicant has provided an adequate description of the control of nonconforming IROFS element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(15) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.16 Corrective Actions

Section 16.0, "Corrective Action," of the QAPD (USEC, 2006c) states that conditions adverse to quality will be identified and corrected promptly. For significant conditions adverse to quality, the cause of the condition will be determined and corrective action will be taken to preclude recurrence. These actions will be documented, reported to appropriate levels of management, and follow-up action taken to verify implementation of the corrective action.

The staff finds that the applicant has provided an adequate description of the corrective actions element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(16) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.17 QA Records

Section 17.0, "Quality Assurance Records," of the QAPD (USEC, 2006c) describes the QA records system for the identification, retention, retrieval, and maintenance of records that furnish documentary evidence of the control of quality for IROFS. Requirements for lifetime and non-permanent records will be identified. Lifetime and ASME NQA-1 record custodianship responsibility will be specified. The storage facilities for hard copy and microfilm lifetime records will meet the requirements of ANSI/ASME NQA-1, Supplement 17S-1, "Supplementary Requirements for QA Records," Section 4.4 (ANSI/ASME, 1994). For electronic records storage, back-ups or duplicate files will be generated.

The staff finds that the applicant has provided an adequate description of the QA records element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(17) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.18 Audits

Section 18.0, “Audits,” of the QAPD (USEC, 2006c) describes planning and scheduling of internal and external audits that verify compliance with the QA program and to determine its effectiveness. Responsibilities and procedures will be identified for assessing, auditing, documenting, and reviewing results and for designating management levels to review audit results, and provisions will be made for review of audit findings, follow-up action, and documentation. The applicant will perform audits and assessments in order to ensure that comprehensive program oversight is performed at least once every three years. External audits of QL-1 supplier’s will be performed before placement of the supplier on the approved supplier list, and follow-up audits will be conducted at a frequency commensurate with the status and performance of the activity based on annual evaluations of the QL-1 supplier’s performance.

The staff finds that the applicant has provided an adequate description of the audits element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(18) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.3.8.19 QAPD Changes

Section 19.0, “Provisions for Changes,” of the QAPD (USEC, 2006c) states that QAPD changes may be initiated by events such as reorganizations, revised activities, lessons learned, changes to applicable regulations, and process changes. QAPD changes will be governed by approved procedures and will be controlled in accordance with 10 CFR 70.72, “Facility changes and change process.”

The staff finds that the applicant has provided an adequate description of the QAPD changes element. The information provided by the applicant, as described above, meets the guidance in Section 11.4.3.8(19) of NUREG-1520 (NRC, 2002), and is therefore acceptable.

11.4 EVALUATION FINDINGS

11.4.1 CM

NRC staff has reviewed the CM function for the ACP in accordance with the regulatory acceptance criteria of NUREG-1520 (NRC, 2002). The staff’s evaluation found that the applicant’s description of the overall CM program appropriately covered CM policy, design requirements, document control, change control, assessments, and design verification. Based on this evaluation, NRC staff finds the applicant’s CM program acceptable.

The applicant has suitably and acceptably described its implementation strategy for a CM program that meets the requirements of 10 CFR 70.72 and provides adequate assurance that facility changes are identified and controlled. IROFS are required to be identified and documented, organizational responsibilities are identified and defined, and administrative controls, policies, and procedures will be established to maintain the design configuration of the facility. Management-level policies and procedures are described that will provide the assurance of consistency among design basis, design requirements, physical configuration, and facility documentation. These policies and procedures include requirements to perform an analysis and independent safety review of any proposed activity involving IROFS, which ensures that consistency is maintained for new activities or for those changes to existing

activities involving licensed material. The applicant's management measures will include the following elements:

CM Policy

The applicant will develop and implement policies, procedures, organizational structure, and personnel responsibilities necessary to effectively implement its CM program.

Design Requirements

The applicant's design requirements and design bases will be documented and will be appropriately supported by analyses. All documentation will be maintained current and consistent with the physical configuration of the facility.

Document Control

Documents, including drawings, will be appropriately controlled, stored, and maintained. Drawings and related documents captured by the document control system will include those documents necessary to adequately describe IROFS.

Change Control

The applicant will achieve and maintain strict consistency among design requirements, documentation, and the physical configuration of the facility. The applicant will develop and implement the policies, procedures, organizational structure, and personnel responsibilities necessary to ensure consistency in design and design bases. These requirements include adequate analysis, review, approval, and implementation of identified changes to IROFS.

Assessments

The applicant will conduct both initial and periodic assessments of its CM program, to assure its effectiveness in meeting its goals and to correct deficiencies.

11.4.2 Maintenance

The applicant ensures the maintenance of IROFS. The applicant's maintenance program implementation strategy adequately addresses the basic elements required to ensure the availability and reliability of IROFS. These elements include corrective maintenance, PM, functional testing, equipment calibration, and work control for maintenance activities. The applicant's maintenance program will provide assurance that equipment performance will be adequately monitored and assessed, using surveillance testing, functional testing, and maintenance records as a means to monitor performance. The surveillance/monitoring, PM, and functional testing activities described by the applicant provide assurance that the IROFS identified in the ISA Summary (USEC, 2006a) will be available and reliable to prevent or mitigate accidents.

The applicant's maintenance program will be based on the use of approved procedures. It will employ work control methods that properly consider personnel safety, awareness of facility operations, QA, and the requirements of corrective maintenance. The ISA Summary (USEC,

2006a) will be used to identify IROFS requiring maintenance, and the determination of the initial PM intervals will be based on industry experience, with due consideration to the applicant's equipment reliability goals. The maintenance program will provide for training that emphasizes the importance of IROFS identified in the ISA Summary (USEC, 2006a), as well as regulations, codes, and personnel safety.

Based on the above, the staff concludes that the applicant's maintenance functions meet the requirements of Part 70, and provide assurance of protecting the health and safety of workers, the public, and the environment.

11.4.3 Training and Qualifications

Based on review of the applicant's strategy for development and implementation with respect to training and qualification and comparison of these requirements to the review acceptance criteria guidance in NUREG-1520 (NRC, 2002), NRC staff has concluded that the applicant has adequately described and assessed its personnel training and qualification in a manner that: (1) satisfies regulatory requirements, and (2) is consistent with the guidance in NUREG-1520 (NRC, 2002). 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 (i) personnel evaluations. Based on this evaluation, the staff finds the applicant's training program acceptable.

There is assurance, based on NRC's review, that implementation of the applicant's training program, as described in the LA (USEC, 2006b), will result in personnel who are qualified and competent to design, construct, start up, operate, maintain, modify, and decommission the facility safely. The staff concludes that the applicant's plan for personnel training and qualification meets the requirements of Part 70.

11.4.4 Procedures

Based on review of the LA (USEC, 2006b) and comparison of the applicant's implementation strategy to review the acceptance criteria in NUREG-1520 (NRC, 2002), NRC staff has determined that the applicant has described a suitably detailed process for the development, approval, and implementation of procedures. The applicant's strategy for developing implementing procedures appropriately cover IROFS, as well as other items important to protecting the health and safety of workers, the public, and the environment. The staff concludes that the applicant's strategic plan for procedure development will meet the requirements of Part 70.

11.4.5 Audits and Assessments

Based on review of the LA (USEC, 2006b) and comparison of the applicant's detailed plan and process to acceptance criteria in NUREG-1520 (NRC, 2002), NRC staff has concluded that the applicant has adequately described its audit and assessment functions. The staff has reviewed the applicant's implementation process used to describe audits and assessments and the description of its policy directives, plans, and procedural requirements for audits and

assessments, taking into consideration such factors as: (1) the general structure of typical audit and assessment activities; (2) facility procedures to be used to control audit and assessment activities; (3) the use of qualified and independent audit and assessment personnel; (4) the planned use of audit and assessment activities; and (5) documentation planning and implementation of corrective actions, based on the findings and recommendations of audits and assessments. Based on the above, the staff finds that the applicant's implementation plan describing audits and assessments is acceptable.

The staff concludes that the applicant's plan for audits and assessments meets the requirements of Part 70 and provides assurance of protection of the health and safety of workers, the public, and the environment.

11.4.6 Incident Investigations

The applicant will develop processes for incident investigations. The applicant's process, which describes the implementation methodologies with respect to its incident investigation program includes: (1) performing incident investigations of abnormal events that may occur during operation of the facility; (2) determining the root cause(s) and generic implications of the event; (3) recommending corrective actions for ensuring a safe facility and safe facility operations; (4) monitoring and documenting corrective actions to completion; and (5) maintenance of documentation so that "lessons learned" may be applied to future operations at the facility. NRC staff has determined that these developmental and implementational processes adequately follow the review acceptance criteria outlined in NUREG-1520 (NRC, 2002). Accordingly, NRC staff concludes that the applicant's description of the incident investigation process complies with applicable NRC regulations and provides assurance of protection of the health and safety of workers, the public, and the environment.

11.4.7 Records Management

The staff has reviewed the applicant's implementation process concerning its records management system against the acceptance criteria in NUREG-1520 (NRC, 2002). Based on this review, NRC staff has concluded that the system will be effective in collecting, verifying, protecting, and storing information about the facility, its design, operations, and maintenance and the records management system will provide for retrieval of the information in a readable form for the designated lifetimes of the records. The staff verified the adequacy of the applicant's developmental and implementational protocols for using 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 applicant has also provided assurance that any deficiencies in the records management system or its implementation will be detected and corrected in a timely manner. Based on this evaluation, the staff finds that the applicant's description and implementation processes for records management are acceptable

11.4.8 Other QA Elements

Based on its review of the LA (USEC, 2006b), NRC staff has concluded that the applicant has adequately described the application of other QA elements applied to IROFS, management measures, and other safety-related items. NRC concludes that the applicant's strategic

approach in this area adequately address the implementation and application of the QA program that will be used before beginning operations at the facility. The staff has also determined that the applicant provides assurance that personnel performing quality-related activities will perform work in accordance with approved procedures and must demonstrate suitable proficiency in their assigned tasks. Additional conclusions the staff reached through its evaluation of the applicant's QA process, procedures, and methods include the following:

1. The applicant has established, documented, and developed an organizational structure responsible for developing, implementing, and assessing the management measures for providing assurance of safe facility operations, in accordance with the acceptance criteria in Section 11.4 of NUREG-1520 (NRC, 2002).

2. The applicant has established and documented a program to develop and implement QA elements and administrative measures for staffing, performance, assessing findings, and implementing corrective actions.

3. The applicant has developed a process for preparation and control of written plant procedures, including a process for evaluating changes to procedures, IROFS, and tests. The process for review, approval, and documentation of procedures will be implemented and maintained.

4. The applicant will develop and implement a program of surveillance, tests, and inspections that will provide assurance of satisfactory in-service performance of IROFS. Specified standards, acceptance criteria, and testing steps have been described or provided.

5. Periodic independent audits will be conducted to determine the effectiveness of the management measures. Management measures will provide for documentation of audit findings and implementation of corrective actions.

6. Training requirements have been established and documented for assuring that employees are provided with the skills to perform their jobs safely. Management measures have been provided for evaluation of the effectiveness of training against predetermined objectives and criteria.

7. The organizations and personnel responsible for performing QA functions will have the required independence and authority to effectively carry out their QA element functions without undue influence from those directly responsible for process operations.

8. QA elements adequately cover the IROFS, as identified in the ISA Summary (USEC, 2006a), and measures are established to prevent hazards from escalating into higher-risk events or accidents.

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

11.5 REFERENCES

(ANSI/ASME, 1994) American National Standard Institute/American Society of Mechanical Engineers Nuclear Quality Assurance (ANSI/ASME NQA). ANSI/ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications,” 1994.

(ASNT, 1980) American Society for Nondestructive Testing (ASNT). SNT-TC-1A, “American Society for Nondestructive Testing Recommended Practice,” 1980.

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

(USEC, 2006a) USEC Inc. (USEC). “Integrated Safety Analysis Summary for the American Centrifuge Plant in Piketon, Ohio,” Revision 14, August 2006.

(USEC, 2006b) USEC Inc. (USEC). “License Application for the American Centrifuge Plant in Piketon, Ohio,” Revision 18, September 2006.

(USEC, 2006c) USEC Inc. (USEC). “Quality Assurance Program Description for the American Centrifuge Plant in Piketon, Ohio,” Revision 3, August 2006.

12.0 SAFETY EVALUATION REPORT PREPARERS

The individuals and organizations listed below are the principal contributors to the preparation of this Safety Evaluation Report. U.S. Nuclear Regulatory Commission (NRC) staff directed the effort and contributed to the technical evaluations. Staff also used contractor support from the Center for Nuclear Waste Regulatory Analyses (CNWRA) and ICF Consulting in the preparation of this document.

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APPENDIX A INTEGRATED SAFETY ANALYSIS (ISA) AND ISA SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the applicant's Integrated Safety Analysis (ISA) and ISA Summary is documented in Appendix A of this Safety Evaluation Report. Appendix A contains information that has been marked as "Export Controlled Information" by the applicant, and is designated by the NRC as "Official Use Only-DOE/NOFORN."

NRC staff concluded that the applicant provided an acceptable ISA Summary for the proposed facility that will meet the applicable 10 CFR Part 70 requirements. The ISA Summary used the What-If/Checklist procedure to determine credible accident sequences. The ISA Summary provides estimates of the likelihood and consequences of each accident sequence, and provides sufficient information to determine whether adequate engineering or administrative controls are identified for each accident sequence. The ISA Summary describes items relied on for safety, management measures, likelihoods and consequences for higher-risk accident sequences, and acceptable methods for achieving the performance requirements in 10 CFR Part 70, Subpart H, "Additional Requirements for Certain Licensees Authorized to Possess a Critical Mass of Special Nuclear Material."

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APPENDIX B ACCIDENT ANALYSIS FOR THE PROPOSED AMERICAN CENTRIFUGE PLANT (ACP)

The U.S. Nuclear Regulatory Commission (NRC) staff independently evaluated the consequences of a set of potential accident sequences identified in the applicant's Integrated Safety Analysis (ISA) Summary. This evaluation is documented in Appendix B of this Safety Evaluation Report. Appendix B contains information that has been marked as "Export Controlled Information" by the applicant, and is designated by the NRC as "Official Use Only-DOE/NOFORN."

A summary of the health effects associated with the five potential accident sequences analyzed by staff is presented in Appendix B. The accident consequences vary in magnitude, and include accidents initiated by human error and equipment failure. The most significant consequences are associated with the release of uranium hexafluoride and nuclear criticality. The proposed American Centrifuge Plant (ACP) design reduces the risk (likelihood) of the accident by identifying items relied on for safety (IROFS), and defense-in-depth features. NRC staff independently verified the accident analysis by performing confirmatory hand calculations and computer simulations. NRC staff concluded that through the combination of plant design, passive and active engineered IROFS, administrative IROFS, and defense-in-depth features, the proposed ACP will pose an acceptably low safety risk to workers, public, and the environment. As a result, the staff determined that the applicant meets the requirements to operate the proposed facility under 10 CFR Part 70.

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APPENDIX C NUCLEAR CRITICALITY SAFETY

The U.S. Nuclear Regulatory Commission (NRC) staff independently evaluated the consequences of a set of potential nuclear criticality accident sequences identified in the applicant's Integrated Safety Analysis (ISA) Summary. This evaluation is documented in Chapter 5 and Appendix C of this Safety Evaluation Report. Appendix C contains information that has been marked as "Export Controlled Information" by the applicant, and is designated by the NRC as "Official Use Only-DOE/NOFORN."

A summary of the potential accident sequences analyzed by staff is presented in Appendix C. The accident consequences vary in magnitude, and include accidents initiated by human error and equipment failure. The proposed ACP design reduces the risk (likelihood) of the accident by identifying items relied on for safety (IROFS), double contingency, and defense-in-depth features. NRC staff reviewed the applicant's analysis for various enrichment levels and compared the values to current NRC guidance and the validation reports. The Emergency Plan was also reviewed in order to determine that the procedures were acceptable from a nuclear criticality safety standpoint. NRC staff concluded that through the combination of plant design, passive and active engineered IROFS, administrative IROFS, and defense-in-depth features, the proposed ACP will pose an acceptably low safety risk to workers, public, and the environment. As a result, the staff determined that the applicant meets the requirements to operate the proposed facility under 10 CFR Part 70.

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APPENDIX D FIRE SAFETY

The U.S. Nuclear Regulatory Commission (NRC) staff independently evaluated the consequences of a set of potential fire and explosion related accident sequences identified in the applicant's Integrated Safety Analysis (ISA) Summary. This evaluation is documented in Chapter 7 and Appendix D of this Safety Evaluation Report. Appendix D contains information that has been marked as "Export Controlled Information" by the applicant, and is designated by the NRC as "Official Use Only-DOE/NOFORN."

NRC staff reviewed combustible loading, flammable material, and a selected set of fire related accident sequences. A list of items relied on for safety (IROFS) related to fire safety is presented in Appendix D. The proposed American Centrifuge Plant (ACP) design reduces the risk (likelihood) of the accident by identifying items relied on for safety (IROFS) and defense-in-depth features. NRC staff concluded that through the combination of plant design, passive and active engineered IROFS, administrative IROFS, and defense-in-depth features, fire related accidents at the proposed ACP will pose an acceptably low safety risk to workers, public, and the environment. As a result, the staff determined that the applicant meets the requirements to operate the proposed facility under 10 CFR Part 70.

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APPENDIX E ELECTRICAL SYSTEM AND INSTRUMENTATION AND CONTROL

The U.S. Nuclear Regulatory Commission (NRC) staff independently evaluated the consequences of a set of potential fire and explosion related accident sequences identified in the applicant's Integrated Safety Analysis (ISA) Summary. This evaluation is documented in Appendix E of this Safety Evaluation Report. Appendix E contains information that has been marked as "Export Controlled Information" by the applicant, and is designated by the NRC as "Official Use Only-DOE/NOFORN."

NRC staff reviewed electrical system and instrumentation and controls (I&C) related accident sequences. This included reviewing the baseline design criteria, general design criteria, applicable Institute of Electrical and Electronics Engineers standards, and applicable American National Standards Institute standards. The proposed American Centrifuge Plant (ACP) design reduces the risk (likelihood) of the accident by identifying items relied on for safety (IROFS), and defense-in-depth features. NRC staff concluded that through the combination of plant design, passive and active engineered IROFS, administrative IROFS, and defense-in-depth features, electrical system and I&C related accidents at the proposed ACP will pose an acceptably low safety risk to workers, public, and the environment. As a result, the staff determined that the applicant meets the requirements to operate the proposed facility under 10 CFR Part 70.

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APPENDIX F STRUCTURAL AND GEOTECHNICAL DESIGN

The U.S. Nuclear Regulatory Commission (NRC) staff independently evaluated the protection of building structures against natural phenomena in the applicant's Integrated Safety Analysis (ISA) Summary. This evaluation is documented in Appendix F of this Safety Evaluation Report. Appendix F contains information that has been marked as "Export Controlled Information" by the applicant, and is designated by the NRC as "Official Use Only-DOE/NOFORN."

NRC staff reviewed structural and geotechnical design of the proposed facility. This included reviewing the building designs, general design criteria, loading conditions, and applicable American Society of Civil Engineers standards. The proposed American Centrifuge Plant (ACP) design reduces the risk (likelihood) of the accident by identifying items relied on for safety (IROFS), and defense-in-depth features. NRC staff concluded that through the combination of plant design, passive and active engineered IROFS, administrative IROFS, and defense-in-depth features, adequate protection of building structures against natural phenomena will pose an acceptably low safety risk to workers, public, and the environment. As a result, the staff determined that the applicant meets the requirements to operate the proposed facility under 10 CFR Part 70.

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APPENDIX G HUMAN FACTORS

The U.S. Nuclear Regulatory Commission (NRC) staff independently evaluated the human factors sections identified in the applicant's Integrated Safety Analysis (ISA) Summary. This evaluation is documented in Appendix G of this Safety Evaluation Report. Appendix G contains information that has been marked as "Export Controlled Information" by the applicant, and is designated by the NRC as "Official Use Only-DOE/NOFORN."

NRC staff reviewed human factors related accident sequences. This included reviewing the human-system interfaces (e.g., alarms, controls, displays), general design criteria, and applicable Institute of Electrical and Electronics Engineers standards. The proposed American Centrifuge Plant (ACP) design reduces the risk (likelihood) of the accident by identifying items relied on for safety (IROFS), and defense-in-depth features. NRC staff concluded that through the combination of the ACP design, passive and active engineered IROFS, administrative IROFS, and defense-in-depth features, human factor related accidents at the proposed ACP will pose an acceptably low safety risk to workers, public, and the environment.

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APPENDIX H MATERIAL CONTROL AND ACCOUNTING

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the applicant's nuclear material control and accountability (MC&A) program is documented in Appendix H of this Safety Evaluation Report. Appendix H contains information that has been marked as "Proprietary Information" and "Export Control Information" by the applicant, pursuant to 10 CFR 2.390 and 10 CFR 810, respectively.

The NRC staff concluded that the applicant provided an acceptable Fundamental Nuclear Material Control Plan (FNMCP) for the proposed facility that will meet the applicable 10 CFR Part 74 requirements. The FNMCP describes acceptable methods for achieving the performance objectives in 10 CFR 74.33(a) and the system capabilities of 10 CFR 74.33(c). As a result, the staff determined that the applicant meets the requirements in the area of MC&A to operate the proposed facility under Part 74.

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APPENDIX I PHYSICAL PROTECTION

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the applicant's Physical Security Plan (PSP) for the protection of Special Nuclear Materials (SNM) of Low Strategic Significance (LSS) is documented in Appendix I of this Safety Evaluation Report. Appendix I contains information that has been marked as "Proprietary Information" by the applicant, pursuant to 10 CFR 2.390.

NRC staff reviewed the applicant's PSP. The methods and procedures outlined in the PSP satisfy the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.67 and 73.71. The PSP for the facility is acceptable and meets requirements for physical protection of SNM-LSS.

The protection of classified matter is described in the Security Plan for the Protection of Classified Matter. Evaluation of this plan is discussed in Chapter 1 of this Safety Evaluation Report.

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APPENDIX J PHYSICAL SECURITY OF THE TRANSPORTATION OF SPECIAL NUCLEAR MATERIAL OF LOW STRATEGIC SIGNIFICANCE

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the applicant's Physical Security Plan (PSP) for the Transportation of Special Nuclear Material (SNM) of Low Strategic Significance (LSS) is documented in Appendix J of this Safety Evaluation Report. Appendix J contains information that has been marked as "Proprietary Information" by the applicant, pursuant to 10 CFR 2.390.

NRC staff reviewed the applicant's PSP for SNM-LSS shipments originating from, or arriving at, the facility. The approaches and procedures outlined in the PSP satisfy the performance objectives, systems capabilities, and event and advance notification requirements specified in 10 CFR 73.67(c), 73.67(g)(1)-(5), 73.71, 73.73, and 73.74. NRC staff has concluded that the facility PSP is acceptable and meets requirements for physical protection of SNM-LSS in transit.