
January 17, 2006
AET 06-0008

Mr. Jack R. Strosnider
Director, Office of Nuclear Material Safety and Safeguards
Attention: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

**American Centrifuge Plant**
**Docket Number 70-7004**
**Submittal of Planned Changes for the License Application and Supporting Documents for the American Centrifuge Plant (TAC Nos. L32306, L32307, and L32308)**

Dear Mr. Strosnider:

Pursuant to Reference 1, USEC Inc. (USEC) hereby submits planned changes for the License Application and supporting documents for the American Centrifuge Plant as Enclosure 1. The planned changes will be finalized and submitted to the NRC in the next revision of the License Application and supporting documents.

In addition to the changes noted in Reference 1, additional changes have been made to Chapters 2.0 and 4.0 of the License Application and Quality Assurance Program Description to incorporate USEC's latest organizational changes. Also, in Chapters 1.0, 3.0, and 4.0 of the License Application there are planned changes to incorporate clarifications discussed with the NRC staff in the topical areas of radiation protection and initial conditions. Revision bars in the right-hand margin depict changes from the previous version submitted to the U.S. Nuclear Regulatory Commission.

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NMSS01

If you have any questions regarding this matter, please contact Peter J. Miner at (301) 564-3470.

Sincerely,

Steven A. Toelle
Director, Nuclear Regulatory Affairs

cc: Y. Faraz, NRC HQ
B. Smith, NRC HQ
C. Tripp, NRC HQ

Enclosure: As Stated

Reference:

1. USEC letter AET 06-0007 from S.A. Toelle (USEC) to J.R. Strosnider (NRC) regarding Submittal of Additional Information Related to Nuclear Criticality Safety for the American Centrifuge Plant (TAC Nos. L32306, L32307, and L32308) – Export Controlled Information, dated January 13, 2006.

10 CFR 70.17 allows the Commission, upon application of any interested person or upon its own initiative, to grant such exemptions from the requirements of the regulations in this part 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. The requested exemption is authorized by law because there is no statutory provision prohibiting the grant of the exemption. The requested exemption will not endanger life or property or the common defense and security and is otherwise in the public interest for the reasons discussed below.

Transportation, handling and storage of solid $UF_6$ filled cylinders are doubly contingent. Double contingency is established by multiple controls that limit the likelihood for a solid product cylinder to be breached during transportation, handling or storage, and the likelihood for a breach to not be identified and repaired before sufficient moderation results in a criticality. Moderation control of $UF_6$ filled cylinders is maintained by ensuring cylinder integrity through periodic cylinder inspections. If a $UF_6$ filled cylinder is found to be breached, the cylinder is covered within 24-hours after discovery to reduce the potential accumulation of moderating material, i.e., rainwater. This time limit ensures a corresponding heavy rainfall will not result in accumulation of sufficient amounts of water to cause a criticality. Damaged cylinders are repaired as necessary and emptied. $UF_6$ cylinders are uniquely identified and their design requirements are controlled to further ensure cylinder integrity and reliability (i.e., $UF_6$ cylinders are QL-1 components and are controlled in accordance with the Quality Assurance Program Description), and USEC implements onsite cylinder handling practices (i.e., requiring the use of approved equipment in accordance with approved procedures), which reduces the likelihood that a solid $UF_6$ cylinder would be breached. These requirements are established as items relied on for safety to ensure the health and safety of the public and workers.

The $UF_6$ cylinders stored in storage yards are not covered by a criticality monitoring system unless those cylinders contain licensed material greater than 5.0 weight percent $^{235}U$. NCS evaluation of product cylinders of any size, configured in infinite planar arrays, containing material enriched up to 5.25 weight percent $^{235}U$, has concluded that subcritical conditions are maintained. The ACP ISA has concluded that cylinders containing licensed material less than or equal to 5.0 weight percent $^{235}U$ cannot be involved in a criticality accident sequence that has a probability of occurrence that exceeds $5 \times 10^{-6}$/year.

The frequencies of criticality events in the cylinder yards have been decreased to the Highly Unlikely range ($<10^{-5}$/year) through the establishment of preventive controls established by the ISA in accordance 10 CFR 70.62. Considering the conservatism of the ISA methodology in developing the unmitigated frequency and actual historical data related to cylinder operations, the frequency values could be reduced further. This additional reduction considers the fact that during 50 years of GDP operations, only one cylinder breach has occurred due to

## 1.4 Application Codes, Standards, and Regulatory Guidance

The ACP utilizes a number of the facilities that were originally constructed to support the GCEP and the GDP. The buildings/facilities were designed and constructed according to DOE requirements and/or nationally accepted codes and standards applicable at the time. Many of those codes and standards were earlier versions of current codes and standards that are utilized today for new construction. The codes and standards of record will be verified and documented during the ACP design verification process discussed in Section 11.1.6 of this license application. Any deviations from the codes and standards of record will be evaluated and documented in accordance with the Configuration Management Program as described in Section 11.1 of this license application. New buildings/facilities will meet the codes and standards applicable at the time the facility is designed and constructed as stated in plant design criteria. Modifications to existing buildings and/or facilities will be evaluated to determine if there is a safety benefit from applying current codes and standards and justification will be documented if current codes and standards are not applied.

The following sub-sections list the various industry codes, standards, and regulatory guidance documents that have been referenced in this license application. The extent to which USEC satisfies each code, standard, and guidance document is identified individually in the sub-sections.

To establish definitive guidance for the design of the American Centrifuge Plant, USEC proposes that the license be conditioned as follows:

> USEC will obtain prior NRC review and approval before deleting or modifying the commitment to any code or standard contained in Section 1.4 of the License Application.

### 1.4.1 American National Standards Institute/American Nuclear Society

- ANSI/ANS 3.1-1987, *Selection, Qualification, and Training of Personnel for Nuclear Power Plants*

  USEC utilizes the provisions contained in 4.3.3, 4.4.5, and 4.5.3.2 of this standard to develop qualifications of radiation protection personnel.

  For the reference to this standard, see Section 4.5.4 of this license application.

- ANSI/ANS 3.2-1994, *Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants*

  USEC utilizes the provisions contained in Appendix A.6, paragraph (a) of this standard.

  For the reference to this standard, see Section 11.4.2.1 of this license application.

For references to this standard, see Sections 11.3.1.1.2, 11.3.1.4, and 11.3.1.9 of this license application.

- ANSI/ANS-8.21-1995, *American National Standard for Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*

  USEC satisfies the provisions of this standard.

  For references to this standard, see Section 5.4.1 of this license application.

- ANSI/ANS-8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response*

  USEC satisfies the provisions of this standard.

  For references to this standard, see Section 5.4.4 of this license application and Section 2.2.4 of the Emergency Plan for the American Centrifuge Plant.

### 1.4.2 American National Standards Institute

- ANSI N13.6-1999, *Practice for Occupational Radiation Exposure Records Systems*

  USEC utilizes the provisions contained in Sections 4, 5, 6, and 7 of this standard for determining radiation protection exposure records.

  For the reference to this standard, see Section 4.8.5 of this license application.

- ANSI N323-1978, *Radiation Protection Instrumentation Test and Calibration*

  USEC satisfies the provisions of this standard, except for Sections 4.6 and 5.1.3.

  For the reference to this standard, see Section 4.8.4 of this license application.

- ANSI N14.1-2001, *Nuclear Materials - Uranium Hexafluoride - Packaging for Transport*

  USEC satisfies the provisions of this standard, except for portions superseded by Federal Regulations with the following exceptions/clarifications:

  A. Cylinders/Valves: Cylinders and valves that are already owned and operated by the United States Enrichment Corporation GDP's and were not purchased to this ANSI N14.1-2001 specifications, but were manufactured to meet previous committed versions of the ANSI standards or specifications at the time only satisfy ANSI N14.1-2001 Sections 4, 5, 6.2.2 to 6.3.5, 7 and 8. Cylinders of this type may be subsequently transferred to the ACP.

### 1.4.7 Nuclear Regulatory Commission Guidance

- Regulatory Guide 1.59, Revision 2, *Design Basis Floods for Nuclear Power Plants*

  USEC satisfies the provisions of this Regulatory Guide (RG) to the extent applicable to a Part 70 licensee.

  For references to this standard, see Sections 1.3.4.3 and 1.3.4.3.2 of this license application.

- Regulatory Guide 3.67, Revision 0, *Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities*

  USEC utilized the provisions of this RG as guidance for DOE reservation Emergency Plan.

  For references to this RG, see Sections 8.1 and 8.2 of this license application.

- Regulatory Guide 3.71, Revision 0, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*

  This RG endorses ANSI/ANS-8 standards. USEC commits to ANSI/ANS-8.1-1983, ANSI/ANS-8.19-1996, and ANSI/ANS-8.20-1991 as described above.

  For the reference to this RG, see Section 5.5 of this license application.

- Regulatory Guide 8.13, Revision 2, *Instructions Concerning Prenatal Radiation Exposure*

  USEC satisfies the provisions of this RG.

  For the reference to this RG, see Section 4.1.1 of this license application.

- Regulatory Guide 8.25, Revision 2, *Air Sampling in the Workplace*

  USEC satisfies the provisions contained in Sections 1, 2, 5, and 6 of this RG.

  For the reference to this RG, see Section 4.7.5 of this license application.

- Regulatory Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses*

  USEC satisfies the provisions contained in Section 7 of this RG.

  For the reference to this RG, see Section 4.7.3 of this license application.

13. United States Oceanic and Atmospheric Administration, National Climactic Data Center, Asheville, NC, Waverly and Piketon Ohio Weather Stations data from 1930 through 2002, and Website: (http://nndc.noaa.gov/onlinstore.html) [NOAA 2003a]

14. Regulatory Guide 1.59, Revision 2, *Design Basis Floods for Nuclear Power Plants*

15. ORO-EP-123, "Preliminary Safety Analysis Report for the Gas Centrifuge Enrichment Plant," Portsmouth, OH, U.S. Department of Energy Oak Ridge Operations Office, July 1980

16. ORO-EP-120, "Seismic Design Criteria for the Gas Centrifuge Enrichment Plant – GCEP," U.S. Department of Energy Oak Ridge Operations Office, Office of the Deputy Manager for Enrichment Expansion Projects, Oak Ridge, Tennessee, December 1978

17. Beavers, J. E., Manrod, W. E., and Stoddart, W. C., K/BD-1025/R1, "Recommended Seismic Hazards Levels for Oak Ridge, Tennessee; Paducah, Kentucky; Fernald, Ohio; and Portsmouth, Ohio," U.S. Department of Energy Reservations, Union Carbide Corporation – Nuclear Division, Oak Ridge, TN, 37830, December 1982

18. "Gas Centrifuge Enrichment Plant, Portsmouth, Ohio, Geotechnical Investigation," Law Engineering Testing Company, Project MK7502, Contract No. EY-77-C-05-5614, April 1978

19. USEC-651, "The $UF_6$ Manual – Good Handling Practices for Uranium Hexafluoride," Revision 8, January 1999

20. ASTM C1052, *Standard Practice for Bulk Sampling of Liquid Uranium Hexafluoride*, 2001

21. Report of Site-Specific Seismic Study, USEC American Centrifuge, Piketon, Ohio, Prepared by Engineering Consulting Services, LLC, ECS Project No. 14-03046, January 2006

22. U.S. Nuclear Regulatory Commission, Environmental Assessment of the USEC American Centrifuge Lead Cascade Facility, January 2004

### 2.1.1 Vice President, American Centrifuge

The Vice President, American Centrifuge located at headquarters, reports to the Senior Vice President. The Vice President, American Centrifuge has overall responsibility for safe operation of the ACP and has shutdown and stop work authority for the ACP. If such 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, six years nuclear experience, and ten years of management experience, which may be concurrent with the nuclear experience.

The USEC Board of Directors appoints the Vice President, American Centrifuge.

### 2.1.2 Director, Regulatory and Quality Assurance

The Director, Regulatory and Quality Assurance, located at headquarters, reports to the Vice President, American Centrifuge.

This position has responsibility for the management of regulatory and quality assurance functions and the ACP policy system. This individual is the primary day-to-day interface with the NRC and has overall responsibility for management of activities related to license requirements for the ACP. Although this individual works closely with the Director, American Centrifuge Plant and key plant personnel, he/she is independent from production, plant operating cost, and production schedule concerns, and has the authority to stop work if there is 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.

This position has, as a minimum, a bachelor's degree in engineering or physical sciences or equivalent technical experience, and six years of nuclear experience, and six years of management experience, which may be concurrent with the nuclear experience.

The Vice President, American Centrifuge appoints the Director, Regulatory and Quality Assurance.

#### 2.1.2.1 Regulatory Manager

The Regulatory Manager, located at the ACP, reports to the Director, Regulatory and Quality Assurance.

The Regulatory Manager is responsible for regulatory oversight functions, environmental compliance, and commitment management. The Regulatory Manager, as delegated by the Director, Regulatory and Quality Assurance, and Director, American Centrifuge Plant, maintains the day-to-day interface with NRC representatives on matters of regulatory compliance. The individual has responsibility for managing the plant change process and ensuring the plant

change reporting requirements are met. The Regulatory Manager is also responsible for 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 to facilitate compliance with implementing policies and procedures.

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 four years of nuclear experience.

The Director, Regulatory and Quality Assurance appoints the Regulatory Manager, with concurrence from the Director, American Centrifuge Plant.

### 2.1.2.2 Quality Assurance Manager

The QA Manager, located at the ACP, reports to the Director, Regulatory and Quality Assurance.

The QA Manager has the responsibility to exercise oversight of procurement, refurbishment, construction, start-up, and plant operations to ensure that the health and safety of the public and workers are adequately protected; to ensure compliance with safety, safeguards, and quality requirements; and to ensure implementation of the Quality Assurance Program Description (QAPD) for the ACP, policies, procedures, and management expectations.

The QA Manager has direct access to the Vice President, American Centrifuge for quality assurance matters and has shutdown and stop work authority, when necessary, to ensure protection of public and worker health and safety; provide for common defense and security; and to ensure regulatory and quality compliance. If such authority is exercised, the QA Manager must concur with restart of shutdown operations. The QA Manager has access to information at the plant related to safety, safeguards, and quality. This manager interacts directly with the Director, American Centrifuge Plant, other managers, and key ACP personnel, and participates (as desired) in any evaluations or discussions related to safety, safeguards, and quality. The QA Manager informs the Director, American Centrifuge Plant and the Director, Regulatory and Quality Assurance about safety, safeguards, and quality issues and compliance.

The QA Manager provides independent oversight and assessment to ensure that the health and safety of the public and workers are adequately protected; to ensure compliance with safety, safeguards, and quality requirements; and to ensure implementation of policies, procedures and management expectations.

The QA Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years nuclear experience, and four years of management experience in quality assurance; nuclear safety oversight; engineering and technical support; or regulatory affairs, which may be concurrent with the nuclear experience.

The Director, Regulatory and Quality Assurance appoints the QA Manager, with concurrence from the Vice President, American Centrifuge.

### 2.1.3 Director, American Centrifuge Plant

The Director, American Centrifuge Plant, located at the ACP, reports to the Vice President, American Centrifuge.

The Director, American Centrifuge Plant is responsible for the day-to-day safe operation of the plant, compliance with applicable NRC regulatory requirements, and adherence to applicable policies. The Director, American Centrifuge Plant is responsible for the overall safe operation and maintenance of the ACP, including refurbishment/construction, initial start-up, testing, and operation. The Director, American Centrifuge Plant is responsible for training, procedures, engineering, and occupational, environmental, and nuclear safety. The Director, American Centrifuge Plant also has primary responsibility for the interface with NRC inspection personnel on matters of regulatory compliance, and may delegate responsibility for this day-to-day interface to the Regulatory Manager.

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 Director, American Centrifuge Plant has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, six years of nuclear experience, and six years of management experience, which may be concurrent with the nuclear experience.

The Vice President, American Centrifuge appoints the Director, American Centrifuge Plant.

#### 2.1.3.1 Plant Support Manager

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.

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 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 Plant Support Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Director, American Centrifuge Plant appoints the Plant Support Manager, with concurrence from the Vice President, American Centrifuge.

#### 2.1.3.1.1 Fire Safety Manager

The Fire Safety Manager reports to the Plant Support Manager.

The Fire Safety Manager is responsible for fire protection services, including interpretation and application of applicable fire codes and standards and emergency management; and 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 Fire Safety Manager has, as a minimum, a bachelor's degree or equivalent technical experience, four years of fire protection experience, and six months of nuclear experience.

The Plant Support Manager appoints the Fire Safety Manager, with the concurrence of the Director, American Centrifuge Plant.

#### 2.1.3.1.2 Nuclear Materials Control and Accountability Manager

The Nuclear Materials Control and Accountability (NMC&A) Manager reports to the Plant Support Manager.

The NMC&A Manager is responsible for ensuring that an effective NMC&A program is implemented and 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 NMC&A Manager has, as a minimum, a bachelor's degree or equivalent technical experience, and four years NMC&A experience.

The Plant Support Manager appoints the NMC&A Manager, with the concurrence of the Director, American Centrifuge Plant.

### 2.1.3.2 Engineering Manager

The Engineering Manager reports to the Director, American Centrifuge Plant.

The Engineering Manager is responsible for engineering activities in support of operations including projects (i.e., design, fabrication, and construction of plant modifications or additions), system engineering, procurement, construction management, and construction engineering; as well as providing the primary interface with the refurbishment/construction contractor(s), and records management and document control. The Engineering Manager manages the design change process for the ACP.

The Engineering Manager is responsible for the Nuclear Criticality Safety (NCS) Program and for maintaining the Integrated Safety Analysis (ISA) for the ACP.

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. If such authority is exercised, the Engineering Manager must concur with restart of shutdown operations.

The Engineering Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Director, American Centrifuge Plant appoints the Engineering Manager with concurrence from the Vice President, American Centrifuge.

#### 2.1.3.2.1 Nuclear Safety Manager

The Nuclear Safety Manager reports to the Engineering Manager.

The Nuclear Safety Manager is 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. If such authority is exercised, the Nuclear Safety Manager must concur with restart of shutdown operations.

The Nuclear Safety Manager is also responsible for the management of NCS functions, including administering the NCS program. These duties include programmatic oversight of NCS and NCS training.

The Nuclear Safety Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years nuclear experience, including six months at a uranium processing plant.

The Engineering Manager appoints the Nuclear Safety Manager, with the concurrence of the Director, American Centrifuge Plant.

#### 2.1.3.2.1.1 Nuclear Criticality Safety Manager

The NCS Manager reports to the Nuclear Safety Manager.

The position is responsible for the management of NCS functions, including administering the NCS program and conducting assessments of program implementation. These duties include programmatic oversight of NCS and NCS training. 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 NCS Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years nuclear experience, including six months at a uranium processing facility where NCS was practiced.

The Nuclear Safety Manager appoints the Nuclear Criticality Safety Manager, with the concurrence of the Engineering Manager.

### 2.1.3.3 Manager, Enrichment Operations

The Manager, Enrichment Operations reports to the Director, American Centrifuge Plant.

The Manager, Enrichment Operations is responsible for the day-to-day production activities at the ACP including production support, operations, and maintenance.

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 Manager, Enrichment Operations has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Manager, Enrichment Operations is appointed by the Director, American Centrifuge Plant, with concurrence from the Vice President, American Centrifuge.

#### 2.1.3.3.1 Production Support Manager

The Production Support Manager reports to the Manager, Enrichment Operations.

The Production Support Manager is responsible for industrial safety, industrial hygiene, chemical safety, and the Radiation Protection Program; waste management; environmental survey; and training and procedures. 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 Production Support Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Manager, Enrichment Operations appoints the Production Support Manager, with concurrence from the Director, American Centrifuge Plant.

##### 2.1.3.3.1.1 Radiation Protection Manager

The Radiation Protection Manager (RPM) reports to the Production Support Manager.

The RPM is responsible for the Radiation Protection (RP) Program for the plant. The RPM is responsible for providing guidance and direction for establishment and implementation of the RP Program and has the authority to deny access to radiological areas by personnel who do not adhere to radiological protection requirements. The RPM has oversight of radiological protection procedures with the authority to oversee and to maintain the integrity of the RP Program. The RPM has direct access to the Director, American Centrifuge Plant and the Vice President, American Centrifuge for radiation protection matters, and 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 RPM has, as a minimum, a bachelor's degree in engineering, health physics, radiation protection, or the physical sciences or equivalent technical experience, and four years experience in radiation protection, including six months at a uranium processing plant.

The Production Support Manager appoints the RPM, with the concurrence of the Manager, Enrichment Operations.

##### 2.1.3.3.1.2 Training Manager

The Training Manager reports to the Production Support Manager.

The Training Manager is responsible for preparation, presentation, and documentation of employee orientations; and for technical and qualification training program development and

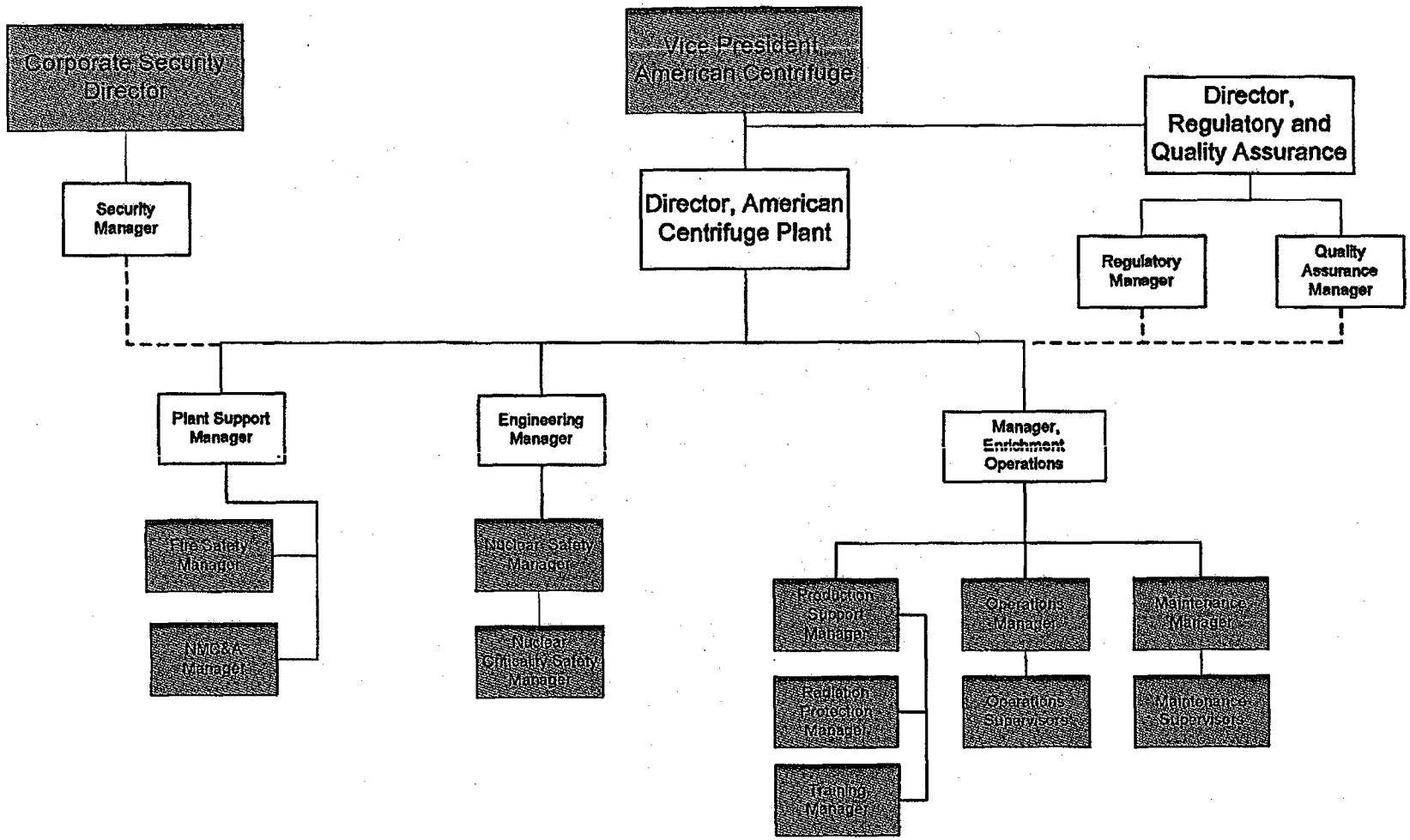


**Figure 2.1-1**
**American Centrifuge Plant Organization Chart**

The HE process is divided into three steps:

- Identification of Initial Conditions and Assumptions;
- Unmitigated Hazard Evaluation; and
- Mitigated Hazard Evaluation.

Initial conditions (ICs) are assumptions that are used to establish a reference baseline for analysis during an evolving design or to clarify a point of analysis that might otherwise be unstated. As such, ICs are normally established and documented prior to or during the HE process.

The Unmitigated HE postulates events that could occur within, or otherwise impact the facility, and assigns event frequencies and event consequences without regard to preventive or mitigative design features or programs, which may be an integral part of facility operations. The unmitigated HE is primarily a qualitative and conservative evaluation of facility hazards to identify those events of most concern to public and worker safety.

If event risk to the public or workers exceeds the 10 CFR 70.61 performance requirements, a more refined analysis may be conducted as part of the Mitigated HE to refine the event frequency and consequences for the event(s) of concern. Alternately, preventive and mitigative features incorporated within the facility and its associated safety programs may be selected and credited as Items Relied on for Safety (IROFS). The Mitigated HE is then developed from the results of the more detailed analysis and/or the crediting of selected preventive and mitigative features to bring the risk of the events within the 10 CFR 70.61 Performance Requirements.

### 3.1.2.3.2.1 Initial Conditions

In order to establish the boundaries of the ISA, the bounding conditions for the ACP must be identified. These boundaries are the operating conditions and limitations under which the ACP is anticipated to operate and in turn are used to establish the ICs credited in the ISA. ICs are the boundary conditions credited in the ISA and are used to establish an analysis reference baseline. ICs are credited during the development of the unmitigated frequencies and event consequences in the ISA. ICs capture assumptions to be used during design evolution or clarify points of analysis that might otherwise be unstated. 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 frequencies or consequences of events. ICs have the potential to impact the results of the hazard analysis. ICs are normally established and documented, prior to, or during the HE process, when events are postulated and evaluated. To preserve the integrity of ICs, they are credited and treated as IROFS.

In general, ICs represent assumptions made in the consequences or probability analyses, or specific passive and active 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 analyses, 2) the combustible materials control program serves to limit the presence of material that could fuel facility fires, and 3) the structural seismic specifications serve to establish minimum structural requirements to reduce the frequency of certain events.

Feed, product, and tails header isolation features serve to limit the amount of licensed material that could be released from the process during a loss of confinement event. This allows the consequence analysis to assume a realistic amount of material at risk. In this instance, the IC credits the active design features to limit inleakage to the entire process.

The combustible materials control program serves to limit the amount of combustibles that could be present in an area where licensed material is located. This reduces the probability that a fire could be initiated or spread and grows in intensity causing a release of licensed material. The IC allows the probability analysis to establish the unmitigated frequency for fire related events. The IC credits the fact that good housekeeping practices will ensure combustible materials are adequately controlled.

Structural seismic specifications state that the process building is designed to withstand a 1,000-year return period seismic event. This precludes or significantly reduces the frequency of building debris from falling on and damaging the operating cascade during a seismic event of this magnitude or less. The IC credits the design of the building in preventing or reducing the frequency of a release occurring as a result of a seismic event. Identifying and crediting certain ICs in this manner is advantageous in that it eliminates the postulation of a release resulting from an event with an unreasonable event frequency (e.g., a release from a 50-year return period seismic tremor).

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 the 10 percent assay limit, are not repeated in the event description of each event.

#### 3.1.2.3.2.2 Unmitigated Hazard Evaluation

Information related to Unmitigated HE is collected and organized in "Hazard Evaluation Tables." These tables are useful as a guide for performing HE, and they provide an effective format for documenting both unmitigated and mitigated HE results. HE Tables are generated to address the non-screened hazards associated with the systems and areas identified during the hazard identification process. The HE Tables may be based on facility sections, systems, activities, or areas, and generally include the following information:

- Event Number and Category;
- Event Description (including location, release mechanism, material at risk, initial conditions specific to the event, and hazard source);
- Cause(s);

The sub-events involve postulating the simultaneous occurrence of the primary event AND the failure upon demand of one or more of the mitigative IROFS. The frequency of failure upon demand of mitigative IROFS was developed in a manner similar to that for assigning preventive values to IROFS described in Section 3.1.2.3.2.4.3.1. Each sub-event is then evaluated in the same manner as that described in Sections 3.1.2.3.2.2, 3.1.2.3.2.3, and 3.1.2.3.2.4. In some cases, the likelihood of the combination of the primary event and the failure of mitigative IROFS fall in the Highly Unlikely frequency range. In these cases, no further evaluation is necessary. In other cases in which the resulting frequency of the primary event in combination with the failure of a mitigative IROFS falls in either the Not Unlikely or the Unlikely frequency range, the consequences of those "combination events" must be shown to be sufficiently low such that the final risk still falls in the "B" risk bin.

#### 3.1.2.3.2.7 Nuclear Criticality Safety Analysis Reports "What if..." Event Analysis

Nuclear Criticality Safety Analysis Reports written for the ACP utilize "What if..." event analysis to ensure that double contingency principle requirements have been adequately satisfied. To ensure that all credible nuclear criticality events have been properly identified in Appendix C of the ISA Summary, a review of the "What if..." events was performed and is documented in Appendix H of the ISA Summary.

### 3.1.3 Management Measures

ACP IROFS are identified in the ISA Summary. Management measures are utilized to maintain the IROFS so that they are available and reliable to perform their safety functions when needed. Management measures are the principal mechanism by which the reliability and availability of each IROFS is ensured. Management Measures are described in Chapter 11.0 of this license application. Any IROFS deficiencies are addressed in accordance with the Corrective Action Program.

## 3.2 Integrated Safety Analysis Summary

An ISA Summary for the ACP (Reference 1) meeting the requirements of 10 CFR 70.65(b) was prepared in accordance with the guidance contained in Chapter 3.0 of NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility* and NUREG-1513, *Integrated Safety Analysis Guidance Document*. The ISA Summary is being submitted for review (separate from this license application).

## 3.3 Items Relied on For Safety Boundary Definition

In order to ensure IROFS are available and reliable, their boundaries must be clearly established. The IROFS boundary determination process relies upon the ISA to identify and define the IROFS and their functions. The boundary determination process then uses the ISA and ACP design documentation to establish and identify what structures, systems, components,

4. 40 CFR Part 68, *Risk Management Programs for Chemical Accidental Release Prevention*, U. S. Environmental Protection Agency, Washington, DC

5. 29 CFR 1910.119, *Process Safety Management (PSM) of Highly Hazardous Chemicals*, Occupational Safety and Health Administration, Washington, DC, 1991

6. 40 CFR 355, *Emergency Planning and Notification*, U. S. Environmental Protection Agency, Washington, DC

7. DOE-HDBK-3010-94, *Airborne Release Fractions and Respirable Fractions for Use with DOE Non-Reactor Nuclear Facilities*, U. S. Department of Energy, Washington, DC, 1994

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

9. Current AIHA ERPGs (2004), http://www.aiha.org/Committees/documents/erpglevels.pdf

10. USEC-02, Application for United States Nuclear Regulatory Commission Certification, Portsmouth Gaseous Diffusion Plant, Safety Analysis Report, Volume 2, Section 4.2

11. R. A. Just, "Report on Toxicological Studies Concerning Exposures to $UF_6$ and $UF_6$ Hydrolysis Products," K/D-5573, Rev. 1, Martin Marietta Energy Systems, Inc., Oak Ridge Gaseous Diffusion Plant, Oak Ridge, TN, July 1984

12. ORO-EP-120, *Seismic Design Criteria for the Gas Centrifuge Enrichment Plant – GCEP*, Department of Energy, Oak Ridge Operations Office, Office of the Deputy Manager for Enrichment Expansion Projects, Oak Ridge, TN, December 1978

13. Report of Site-Specific Seismic Study, USEC American Centrifuge, Piketon, Ohio, Prepared by Engineering Consulting Services, LLC, ECS Project No. 14-03046, January 2006

# 4.0 RADIATION PROTECTION

This chapter describes the American Centrifuge Plant (ACP) Radiation Protection (RP) Program for keeping occupational radiation exposures and radioactive contamination below regulatory limits and as low as reasonably achievable (ALARA). The RP Program addresses the occupational radiation protection requirements set forth in 10 *Code of Federal Regulations* (CFR) Parts 19, 20, and 70. The Radiation Protection Manager (RPM) is responsible for the ACP RP Program. The RPM or designee carries out responsibilities of the RPM described in this chapter.

## 4.1 Radiation Protection Program Implementation

In accordance with 10 CFR 20.1101(c), the RP Program content and implementation is reviewed annually. The RPM is responsible for this annual review and preparation of a report documenting the results of the review. The ALARA Committee then reviews the report. Revisions to the RP Program, if warranted, are initiated and processed by the RPM as part of the annual review process. Any resulting changes to the Radiation Worker Training module are also implemented.

## 4.2 As Low As Reasonably Achievable Program

In accordance with 10 CFR 20.1101, the ACP RP Program is designed to protect personnel entering the ACP from unnecessary exposure to ionizing radiation and radioactive materials. This program is based upon the following principles and is implemented through written procedures.

- Personnel radiation exposures and the release of radioactive effluents shall be maintained in accordance with the ALARA principle.
- No individual shall receive a radiation dose in excess of any regulatory limit.

Responsibility for establishing and ensuring adherence to these principles rests with the Vice President, American Centrifuge. The Director, American Centrifuge Plant has the overall responsibility and authority for the ALARA Program. The RPM is responsible for establishing and implementing the ALARA Program in accordance with written policies and procedures.

### 4.2.1 As Low As Reasonably Achievable Committee

The ALARA Committee is an independent advisory group to the Director, American Centrifuge Plant and the Plant Safety Review Committee on RP issues. It functions to: (1) monitor selected operational RP issues; (2) advise ACP management on RP concerns; and (3) review proposed designs, work practices, selected suggestions, and selected projects with regard to contamination control and/or ALARA.

determines if there are any upward trends in personnel exposure for identified categories of workers and types of operations. The review also identifies any upward trends in effluent releases and contamination levels and determines if exposures, releases, and contamination levels are in accordance with the ALARA concept. Specific areas reviewed include, but are not limited to the following:

- Technologies for selected job tasks;
- Current work practices and completed tasks which have/had contamination control or ALARA concerns;
- Radiation protection violations;
- Lessons learned;
- Trends and resulting impacts on contamination control and/or ALARA; and
- Environmental monitoring reports.

The committee also establishes annual contamination control and exposure goals. Minutes are issued that identify committee members and/or alternates in attendance, agenda items, a summary of decisions made, and action items. Copies are made available to ACP management and the committee members. Recommendations of the ALARA Committee are documented and tracked to completion in the Corrective Action Program.

## 4.3 Organization and Personnel Qualifications

The RPM is responsible for providing guidance and direction for establishment and implementation of the RP Program and has direct access to the Director, American Centrifuge Plant and Vice President, American Centrifuge for radiological control matters. The RPM reports to the Production Support Manager, which provides independence from operations. The RPM and designee are required to have the technical competence and experience to establish RP programs (RPM qualifications are stated in Section 2.1.4.3.1.1) and the management capability to direct the implementation and maintenance of RP programs.

The HP Group reports to the RPM and provides radiological protection support to the plant. HP is independent of the organizations responsible for production. The HP Group is staffed with suitably trained individuals who provide oversight and control of the technical aspects of the program elements that affect RP. There are sufficient HP resources available to support ACP activities.

HP Technicians and their managers perform the functions of assisting and guiding workers in the radiological aspects of the job. HP Technicians and their managers have the responsibility and authority to stop radiological work or mitigate the effect of an activity if they

Continuous HP coverage may be used in lieu of RWPs when approved by the RPM. Qualified HP Technicians are authorized to provide continuous radiological coverage in lieu of an RWP for short duration (less than one shift), non-complex tasks. When continuous HP coverage is used, requirements normally specified on an RWP are communicated to the worker verbally.

## 4.5 Training

Radiological control is provided by controlling access to areas where radioactive material may be encountered and by requiring that each person who enters those areas or facilities receive the appropriate level of radiological worker training. Personnel are trained commensurate with the hazard per 10 CFR Parts 19 and 20. Details concerning Visitor Site Access Orientation and radiological training are provided in Section 11.3.1 of this license application. The Radiological Worker Training Program addresses the requirements of 10 CFR 19.11 and 19.12 and workers' responsibilities under the Radiation Protection Program. The Radiation Worker Training program is described in Section 11.3.1.3 of this license application.

### 4.5.1 Visitor Site Access Orientation

Visitors review basic information related to the site and hazards present at the ACP. Trained radiological workers escort visitors who are granted access to the Restricted Areas.

### 4.5.2 General Employee Radiological Training

General Employee Radiological Training covers the employee's responsibilities for maintaining exposures to radiation and radioactive materials in accordance with the ALARA philosophy.

### 4.5.3 Radiation Worker Training

If a person requires unescorted access to the Restricted Area, radiological worker qualification is required. Radiation Worker Training is a biennial training requirement.

### 4.5.4 Health Physics Technician

HP Technicians are trained and qualified in accordance with an approved qualification standard and training is delivered consistent with applicable training procedures (see Section 11.3). The qualification standard is based on the requirements of American National Standards Institute (ANSI)/American Nuclear Society 3.1, *Selection, Qualification, and Training of Personnel for Nuclear Power Plants*, 1987 Edition. HP Technician training develops the skills necessary to perform assigned work in a competent manner. The training consists of initial, on-the-job, and continuing training.

HP Technician qualification consists of the standardized core course training material, ACP-specific information, and on-the-job training. Passing a final comprehensive written examination is required. The training program ensures personnel are proficient in radiation

HEPA filter systems used to implement ALARA principles and to control worker exposures are tested in accordance with ASME N510-1989. For those systems not designed in accordance with ASME N509-1989, *Nuclear Power Plant Air-Cleaning Units and Components*, ASME N510-1989 is used as testing guidance.

The average air velocity through openings in uranium sampling and handling hoods containing readily dispersible uranium is a minimum of 100 linear feet per minute (lfpm). This velocity is checked at least annually.

If radiological containments are used, when they are in use and have the potential to generate airborne radioactivity, they will be maintained at a negative differential pressure.

### 4.6.2 Respiratory Protection

The Respiratory Protection Program follows the requirements of 29 CFR 1910.134 and 10 CFR Part 20 for use, issuance, training, and qualifications for respirator users. Procedures for respirator usage follow the requirements of 10 CFR 20.1703(c)(4). Records of respirator user training and fit testing are maintained as required by Section 11.7 of this license application. RWPs specify respiratory protection required for radiological protection purposes. Respirator use is considered for activities where an individual may be exposed to soluble uranium that may exceed 0.8 DAC-hours or an intake of 1 milligram (mg) of soluble uranium during a work shift.

Engineering and administrative controls, including access restrictions and the use of specific work practices designed to minimize airborne contamination or loss of contamination control are used to minimize worker internal exposure. When engineering and administrative controls have been applied and the potential for airborne radioactivity still exists, respiratory protection is used to limit internal exposures. Use of respiratory protection is considered under any of the following conditions:

- During entry into posted ARAs;
- During breach of contaminated systems or components;
- During work in areas or on equipment with removable contamination levels greater than 100 times the levels in Table 4.6-1; and
- During work on contaminated surfaces with the potential to generate airborne radioactivity.

In specific situations approved by the RPM, respiratory protection may not be used due to physical limitations, such as heat stress, or the potential for significantly increased external exposure. In such situations, stay time controls to limit intakes are established and continuous workplace airborne monitoring is provided along with expedited analysis of results.

**Table 4.6-1 Contamination Levels**

| Nuclide[a] | Removable (dpm/100 $cm^2$)[b] | Total (Fixed + Removable) (dpm/100 $cm^2$) |
|---|---|---|
| U-natural, $^{235}U$, $^{238}U$, and associated decay products, Transuranics $\leq$ 2 percent by alpha activity, $^{99}Tc$, and beta-gamma emitters | 1,000 | 5,000 |
| Transuranic modified materials containing > 2 percent and < 8 percent transuranics by alpha activity, Th-natural, $^{232}Th$, $^{223}Ra$, $^{224}Ra$, and $^{232}U$ | 200 | 1,000 |
| $^{226}Ra$, $^{228}Ra$, $^{230}Th$, $^{228}Th$, $^{231}Pa$, $^{227}Ac$, $^{125}I$, $^{129}I$, and Transuranics $\geq$ 8 percent by alpha activity | 20 | 200 |

[a] The values in this table apply to radioactive contamination deposited on, but not incorporated into the interior of, the contaminated item. Where contamination by both alpha and beta-gamma-emitting nuclides exists, the levels established for the alpha- and beta-gamma-emitting nuclides apply independently.

[b] The amount of removable radioactive material per 100 square centimeters ($cm^2$) of surface area is determined by swiping the area with a dry filter or soft absorbent paper while applying moderate pressure and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. For objects with a surface area less than 100 $cm^2$, the entire surface is swiped; and the activity per unit area is based on the actual surface area. Except for transuranics $\geq$ 8 percent by alpha activity, $^{228}Ra$, $^{227}Ac$, $^{228}Th$, $^{230}Th$, $^{231}Pa$, and alpha emitters, it is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual contamination is within the levels for removable contamination.

The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 $cm^2$ is less than three times the level specified. For purposes of averaging, any square meter of surface is considered to be above the level G if: (1) from measurements of a representative number of n of sections it is determined that $1/n \, \Sigma_n \, S_i \geq G$, where $S_i$ is the disintegration per minute (dpm)/100 $cm^2$ determined from measurements of section i; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 $cm^2$ area exceeds 3G. (G is defined as the levels listed above.)

radiation measurements made with radiation survey instruments. Self-reading or alarming dosimeters are used for entry into HRAs or Very High Radiation Areas.

If an individual exceeds 50 percent of the ACL during a calendar quarter or the ACL in the calendar year, an evaluation is performed by the RPM for approval by the Director, American Centrifuge Plant. The evaluation is performed to determine the types of activities that may have contributed to the worker's exposure. This may include, but is not limited to, procedural reviews, and review of work practices, work locations, and job assignments. Depending upon the conclusions of the evaluation, the individual may be allowed to continue radiological work; however, work restrictions may be imposed on individuals whose exposure exceeds the ACL.

Approval for continued work is documented in the evaluation, as described in the preceding paragraph, which requires approval by the Director, American Centrifuge Plant. Investigations to determine cause, assess the exposure, and document the results are conducted in accordance with written procedures.

HP determines any unusual trends or exposures during reviews of external dosimetry results. If the external exposure status of an individual is uncertain, the individual is removed from further exposure until HP determines the exposure status and advises management of any special controls or restrictions to be applied.

To comply with the reporting requirements of 10 CFR 20.2206, the site submits personnel monitoring information for the Radiation Exposure Information Reporting System (REIRS) report based on the personnel exposure database. This includes summation of internal and external doses as outlined in Section 7 of Regulatory Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses.*

The occupational exposure received by ACP employees, subcontractors, and visitors must not exceed the 10 CFR Part 20, Subpart C limits. The ACP requires current year exposure history of an occupational worker as required by 10 CFR 20.2104.

Personnel declaring pregnancy are advised to control radiation exposure to an embryo or fetus in accordance with the ALARA principle during the entire gestation period. The ACP complies with the guidelines of Regulatory Guide 8.13, Revision 2, *Instructions Concerning Prenatal Radiation Exposure.*

### 4.7.4 Internal

The chemical characteristics and retention times of soluble uranium processed at the ACP are such that renal toxicity limitations are the limiting conditions for health effects. A bioassay program is employed to confirm the results of radioactive material contamination control and respiratory protection programs. Bioassay results are the primary means of calculating internal doses. Personnel who have the potential to receive intakes resulting in a Committed Effective Dose Equivalent (CEDE) greater than or equal to 0.1 roentgen equivalent man (rem) CEDE in a year or intakes of 1 mg of soluble uranium per week participate in the routine bioassay program.

Personnel submit bioassay samples, such as urine or fecal samples, and participate in *Invivo* monitoring as required by the bioassay program. Table 4.7-2 provides a summary of the bioassay program description and the analytical methods employed. The routine sample submission frequencies and administrative control levels are listed in Table 4.7-3.

Because chemical toxicity is limiting when personnel are exposed to soluble uranium, the uranium action levels have been selected to limit an individual's chronic intake to 10 mg of soluble uranium per week. Personnel participate in follow-up bioassay monitoring when their bioassay results exceed administrative control levels or as determined by HP. Special bioassay studies are performed as necessary and investigations performed when intakes are confirmed or suspected to exceed 1 mg of soluble uranium per week.

The ACP collects "random single void" urine samples from personnel. Isotopic analysis of fecal samples and 24-hour urine sampling are not routinely performed, however, these analyses will be considered when dose assessments exceed 0.5 rem CEDE. Bioassay results are used to assign internal dose. The sensitivities of lung counting systems are not as effective as urinalysis for Class D uranium; lung counting is considered when intake estimates exceed 0.5 rem CEDE.

The CEDE per unit of intake by inhalation from Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,* is used to calculate internal dose.

HP determines unusual trends during reviews of urinalysis results. If bioassay sample results indicate an internal exposure that exceeds action levels or appears uncertain, additional analyses and removal of the individual from further exposure are considered.

### 4.7.5 Airborne Radioactivity

The ACP air sampling program is consistent with the basic requirements of Regulatory Guide 8.25, *Air Sampling in the* Workplace, Sections 1, 2, 5, and 6. Routine general area air sampling is established in areas where airborne radioactivity concentrations may exceed 10 percent of the DAC listed in Table 4.7-4, averaged over 8 hours. Table 4.7-4 also summarizes the airborne radioactivity posting levels. Investigations are performed when airborne radioactivity data indicates personnel exposures exceed 0.8 DAC-hours. Special bioassay sampling is required when air samples exceed 0.8 DAC-hours. Adjustment for respirator use is considered in determining bioassay monitoring.

A combination of low-volume, high-volume, and lapel air samplers are used for job coverage and general area air sampling. Low-volume air samplers are used for routine air sampling and are exchanged at least weekly. Due to radon and radon daughter products, routine air samples are allowed to decay for a minimum of three days.

Air sample data is not used as the primary method to determine internal dose, however the data is used to prompt bioassay monitoring. Only air samples collected in the workers' breathing zone (approximately 30 cm) are considered representative.

The RPM is responsible for maintaining adequate quantities of calibrated radiation detection and measurement instruments.

Radiological portable instruments are calibrated based on specifications derived from applicable vendors manuals and other nationally recognized guidance as appropriate (e.g., National Council on Radiation Protection 112). The standards found in the ANSI N323 (1978) are followed except for Sections 4.6 and 5.1(3). The following requirements apply to all such equipment and instruments:

- Portable radiation detection and measurement instruments are inspected, maintained, and calibrated at least annually or removed from service.
- Instruments are calibrated following any maintenance, modification, or repair deemed likely to affect operation before being returned to service.
- Calibration sources and equipment used for dose rate instruments are within 5 percent (at 2 sigma) of the stated value and have documented traceability links to the NIST. Large area uranium slab sources are certified to 10 percent by NIST. Calibration sources used to calibrate contamination-monitoring equipment are within 20 percent (at 2 sigma) for activity and 10 percent (at 2 sigma) for surface emission rate.
- Portable HP instruments that are in use but do not have a built in automatic functional test feature are source checked daily, or prior to using the instrument if not used on a daily basis. Instruments with the automatic functional test feature that are in use are checked once a week.

### 4.8.5 Records and Reports

Radiological protection records demonstrate the effectiveness of the overall program and document personnel exposure. Records are maintained in the form required by 10 CFR 20.2110 and are retained as required by 10 CFR 20.2101 through 20.2106 according to the Records Management Program as outlined in Section 11.7 of this license application. USEC follows the guidance contained in ANSI N13.6, *Practice for Occupational Radiation Exposure Records Systems*, 1999 Edition, for radiological protection records.

Reports and notifications of RP issues are made pursuant to 10 CFR Part 20, Subpart M; 10 CFR 30.50; 10 CFR 40.60; 10 CFR 70.50; and/or 10 CFR 70.74. Events requiring reporting to the NRC are investigated, tracked in a database, and monitored through completion in accordance with the Corrective Action Program. Details of reporting and notification for ACP incidents are described in Section 11.6 of this license application.

## 4.9 References

1. ASME N509-1989, *Nuclear Power Plant Air-Cleaning Units and Components*

2. ASME N510-1989, *Testing of Nuclear Air-Treatment Systems*

3. ANSI/American Nuclear Society 3.1, *Selection, Qualification, and Training of Personnel for Nuclear Power Plants*, 1987 Edition

4. ANSI N13.6, *Practice for Occupational Radiation Exposure Records Systems*, 1999 Edition

5. ANSI N323-1978, *Radiation Protection Instrumentation Test and Calibration*

6. Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*

7. NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*

8. Regulatory Guide 8.13, Revision 2, *Instructions Concerning Prenatal Radiation Exposure*

9. Regulatory Guide 8.25, *Air Sampling in the Workplace*, Sections 1, 2, 5, and 6

10. Regulatory Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses*, Section 7

# 5.0 NUCLEAR CRITICALITY SAFETY

The American Centrifuge Plant (ACP) possesses large quantities of uranium hexafluoride ($UF_6$) at enrichments of up to 10 weight (wt.) percent uranium-235 ($^{235}U$). The specific authorized uses for each class of U. S. Nuclear Regulatory Commission (NRC)-regulated material are shown in Table 1.2-2 of this license application. USEC Inc. is required to comply with the performance requirements of 10 *Code of Federal Regulations* (CFR) 70.61. 10 CFR 70.61(d) requires that the risk of nuclear criticality accidents be limited by assuring that under normal and credible abnormal conditions, nuclear processes are subcritical, including use of an approved margin of subcriticality for safety. It also requires that preventive controls and measures must be the primary means of protection against nuclear criticality accidents. Accordingly, this chapter summarizes the ACP Nuclear Criticality Safety (NCS) Program.

In accordance with the requirements contained in 10 CFR 70.62, the likelihood and risks of an inadvertent nuclear criticality were evaluated in the Integrated Safety Analysis (ISA). The evaluation considered moderation events, maintenance evolutions, machine upset conditions, and cylinder operations. The ISA effort documented these evaluations in NCS Reports that will in turn form the bases to develop Nuclear Criticality Safety Evaluations (NCSEs) addressing the detailed design. If changes to the ISA are identified during the development of the NCSEs, USEC will revise the ISA, as necessary, to include any new or updated event sequence information, identify additional double contingency controls, or credit additional items relied on for safety (IROFS). The ISA concluded that credible nuclear criticality accident scenarios that could be identified for the ACP were controlled through a combination of administrative and engineered controls in compliance with the performance requirements of 10 CFR 70.61(d). The plant has established a threshold of 1 wt. percent or higher enriched $^{235}U$ and 100 grams (g) or more of $^{235}U$ for determining when an evaluation for NCS considerations of planned operations must be performed. This 100 g $^{235}U$ mass is a minimum of a factor of 10 below the minimum critical mass at 10 percent $^{235}U$ enrichment, regardless of whether the material is non-oily, oily, or heterogeneous for a fully reflected system. Based on this, the value is sufficiently low to use as a threshold limit. In view of this threshold, many of the ACP NCS Program features described in this chapter may not be required to be implemented for operations below the threshold. In this regard, the NCS Program provides the framework for a defense-in-depth philosophy to help ensure the risk of inadvertent criticality is maintained acceptably low. The NCS Program also provides the framework and resources for evaluating plant performance in establishing NCS analyses and controls for the design and operation of a uranium enrichment plant.

Information contained within does not contain Export Controlled Information

Reviewer: DA Hupp
Date: 01/17/06

## 5.1 Management of the Nuclear Criticality Safety Program

### 5.1.1 Program Elements

The NCS Program described in this chapter is implemented by plant procedures. The NCS procedures address plant personnel NCS responsibilities, adherence to NCSE requirements, review and approval of fissile material operations, posting and labeling requirements, response to NCSE violations, and NCS training requirements. Controls and/or barriers that are relied on to prevent inadvertent criticalities are designated as IROFS in the ISA. The NCS Program meets the Baseline Design Criteria (BDC) requirements in 10 CFR 70.64(a) concerning application of the double contingency principle in determining NCS controls and IROFS in the design of new facilities.

- **Surface Density Method** using unit shape factor – This method can be used as an approximation for large arrays of identical units containing solutions and metals. This method determines the spacing and mass of units independent of the number of units. An important feature of the Surface Density Method is that it is equally applicable to more irregular geometries.

When hand calculations are used, the specific methodology employed will be as described in "Nuclear Criticality Safety" by R.A. Kneif, American Nuclear Society, 1991 and subject to a total system reactivity of 0.95 for all credible off normal events.

**Computer Calculations**

For those cases where adequate references are not available, NCS computational analyses are performed, which involve the calculation of $k_{eff}$ to determine whether the system will be subcritical under both normal and credible abnormal process conditions. Computer codes that simulate the behavior of neutrons in a process system or that solve the Boltzmann transport equation are used.

Computer calculations of $k_{eff}$ provide a method to relate analytical models of specific system configurations to experimental data derived from critical experiments. A critical experiment is defined as a system that is intentionally constructed to achieve a self-sustaining neutron chain reaction or criticality. Critical experiments that have specific, well-defined parametric values and are adequately documented are termed benchmark experiments. Computer codes are validated using experimental data from benchmark experiments that, ideally, have geometries and material compositions similar to the systems being modeled.

Validation of the computer code determines its calculational bias or uncertainty as well as the effective margin of subcriticality. The validation involves the modeling of benchmark critical experiments over a range of applicability. Because the $k_{eff}$ value of a critical experiment is essentially 1, the bias of the code is taken to be the deviation of the calculated values of $k_{eff}$ from unity. Statistical analysis is employed to estimate the calculational bias, which includes the uncertainty in the bias and uncertainties due to extensions of the area of applicability, as well as the effective margin of subcriticality. Uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of the experimental data. The validation of the computer code specifically defines the maximum acceptable $k_{eff}$ used to determine subcriticality.

The margin of subcriticality used for the plant results in a $k_{eff}$ upper safety limit that ensures that there is a 95 percent confidence that 99.9 percent of future $k_{eff}$ values less than this limit will be subcritical. A minimum margin of subcriticality of 0.02 in $k_{eff}$ is used to establish the acceptance criteria (i.e., upper safety limit) for criticality calculations for abnormal conditions at 5 percent $^{235}U$ enrichment and below. Above 5 percent $^{235}U$ enrichment, a minimum margin of subcriticality of 0.05 in $k_{eff}$ is used. Abnormal conditions are changes to a controlled parameter that result in a violation of the limit on that parameter. For example, in an operation that relies on maintaining spacing between fissile units, an error that results in the units being closer than the limit would represent an abnormal condition. Similarly, operations that rely on moderation control of $UF_6$ would be in an abnormal condition when the moderation control was lost and operations that rely on control of $^{235}U$ mass would be in an abnormal condition when the mass limit was violated.

The upper safety limit varies with the computer system, codes, cross sections, and materials used in the validation.

The calculation of $k_{eff}$ is accomplished by the use of computer codes that utilize Monte Carlo techniques to determine $k_{eff}$ of a system. Computer models representing the geometrical configuration and material compositions of the system are developed for use within the code. The development of appropriate models must account for or conservatively bound both normal and credible abnormal process conditions.

When NCS is based on computer code calculations of $k_{eff}$, controls and limits are established to ensure that the maximum $k_{eff}$ complies with the applicable code validation for the type of system being evaluated. For example, NCS related IROFS developed during initial license application were developed using reactivity calculations performed on personal computers running the Microsoft Windows XP operating system and validated as described in Reference 11. Generally, these calculations were performed with an upper safety limit of 0.955 up to5 percent $^{235}U$ enrichment; however, specific cases may use a higher or lower limit based on equations from Table 14 of Reference 11. Above 5 percent $^{235}U$ enrichment, a margin of subcriticality of 0.05 will be applied to calculations performed using the personal computers described above with a resulting upper safety limit of 0.925. Reactivity calculations, performed after initial license application, comply with the code validation for the specific system used to perform the calculation.

Scoping and analysis calculations may be performed utilizing various unvalidated computer codes; however, computer calculations of $k_{eff}$ used as the basis for NCS evaluations are confirmed by, or performed using, configuration-controlled codes and cross-section libraries for which documented validations are performed with at least the same degree of conservatism as that presented in Reference 11 and are in accordance with ANSI/ANS-8.1-1998. Calculations are performed using materials of construction and other parameters consistent with the area of applicability described by the relevant validation report. The area of applicability used by Reference 11 covers enrichments from 2 percent to 30 percent $^{235}U$ enrichment with moderation levels from an $H/^{235}U$ of 8 to 1,438 with an average energy group of 151.7 to 220 using the 238-group ENDF/B-V cross section library. Moderating materials from Reference 11 include water and paraffin and reflectors range from bare systems to reflection with water, steel, paraffin, polyethylene, concrete, and lead. Other materials included in the area of applicability from Reference 11 are stainless steel, zirconium, aluminum, fluorine, and oxygen. Extensions to the area of applicability are justified when used. The NRC will be notified in the event an extension to the area of applicability will not adequately encompass the parameters of interest for a specific calculation and a revision to Reference 11 is needed to establish a new area of applicability.

The methodology used in a validation report involves statistical analysis to determine the bias and bias uncertainty for the critical experiments included in the validation. Guidance from NUREG/CR-6698, *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*, is used to perform the validation. The upper safety limit is computed by subtracting the absolute value of the bias, the bias uncertainty, and the minimum margin of subcriticality from unity. Positive bias is not credited. The exact statistical technique used to obtain the bias and bias uncertainty depends on the specific validation report. The techniques used in Reference 11 included the lower tolerance limit or the lower tolerance band for normally distributed data and a non-parametric technique for non-normally distributed data.

The computer codes and cross sections used in performing $k_{eff}$ calculations are maintained in accordance with a configuration control plan. Quarterly, or prior to use, one of the following is performed: a bit-by-bit comparison of the production version of the software (executable modules and data libraries) versus an archived production version; or a comparison of the output from all validation cases versus archived output of all validation cases from the original validation performed when the production version was installed to ensure no changes in the calculated $k_{eff}$ for the validation cases. Changes to the hardware or software are evaluated in accordance with 10 CFR 70.72 change requirements. The System Administrator, a NCS engineer, is responsible for controlling access to the software.

## 5.5 References

1. ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*

2. ANSI/ANS-8.3-1997, *Criticality Accident Alarm System*

3. ANSI/ANS-8.19-1996, *Administrative Practices for Nuclear Criticality Safety*

4. ANSI/ANS-8.21-1995, *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*

5. ARH-600, *Criticality Handbook*, Volumes I, II, and III, Atlantic Richfield Hanford Co. report (1968)

6. LA-3605-0003, Integrated Safety Analysis Summary for the American Centrifuge Plant

7. LA-10860-MS, *Criticality Dimensions of Systems Containing $^{235}U$, $^{239}Pu$, and $^{233}U$*, 1986 Revision

8. NRC Regulatory Guide 3.71, Revision 0, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*

9. NUREG-1513, *Integrated Safety Analysis Guidance Document*

10. NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*

11. WSMS-CRT-03-0093, United States Enrichment Corporation (USEC) PC-SCALE 4.4a Validation (U), Revision 2, November 2005

12. NUREG/CR-6698, *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*, January 2001

# 1.0 INTRODUCTION

The Quality Assurance Program Description (QAPD) described herein applies to the design, fabrication, testing, and operation of the American Centrifuge Plant (ACP) and meets 10 *Code of Federal Regulations* (CFR) 70.64 (a)(1).

The ACP is located in Piketon, Ohio. The QAPD is applied using a graded approach as described in Section 2.0 of this QAPD.

## 1.1 Organization

USEC Inc. (USEC) maintains overall responsibility for design, refurbishment/ construction, start-up, and operations.

Figure 1.1-1 of this plan shows the organization for the ACP.

## 1.2 Design, Refurbishment/Construction, Start-up, and Operations Organization

The Vice President, American Centrifuge has overall responsibility for the design, refurbishment/construction, start-up, and operation of the ACP and reports to the Senior Vice President.

The Vice President, American Centrifuge is responsible for the Quality Assurance (QA) Program and for determining the status, adequacy, and effectiveness of the QAPD. The QA Manager reports to the Director, Regulatory and Quality Assurance and has independent oversight responsibility for implementation of the QAPD. The QA Manager has direct access to the Vice President, American Centrifuge for QA matters.

The Vice President, American Centrifuge has designated the Director, American Centrifuge Plant the responsibility for design, refurbishment/construction, start-up, operation, and associated support activities for the ACP. The Director, American Centrifuge Plant is responsible for the ACP and overall responsibility for implementation of the QAPD. The QAPD is binding on all USEC and contractor personnel involved with the ACP.

The Manager, Enrichment Operations reports to the Director, American Centrifuge Plant, and is responsible for day-to-day production activities at the plant.

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 quality assurance functions and the ACP policy system. This individual is the primary day-to-day interface with the 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.

The QA Manager is responsible for independent oversight of ACP activities covered by this QAPD. This includes maintenance of the QAPD and assessing its effective implementation. This includes the responsibility and authority for:

QL-2 items and services are procured as commercially available in accordance with the criteria in this section and the criteria applicable to commercial grade items and services contained in Section 7.0 of this QAPD. However, commercially available items and services, unlike commercial grade items and services, are not required to be dedicated.

## 5.0 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Activities affecting the availability and/or reliability of IROFS are prescribed by and accomplished in accordance with documented procedures, instructions, and drawings of a type appropriate to the circumstances. These documents include or reference appropriate acceptance criteria for determining that prescribed activities have been satisfactorily accomplished. Standard guidelines for the format, content, and review and approval processes are established.

The QAPD establishes the policy requirements approved by the Vice President, American Centrifuge. Procedures are the second tier of documents that implement the QAPD. Third tier instructions provide specific step-by-step directions when deemed necessary. Procedure and instruction preparation, review, and approval are the responsibility of the applicable manager. The QA organization reviews QA implementing procedures for compliance and consistency with this QAPD. QA review of procedures is performed to ensure that the provisions of this QAPD are effectively incorporated into QA implementing procedures.

Adherence to policy, procedures, and instructions is mandatory. In the case of conflict or error involving a procedure, the activity in question shall be placed in a safe condition and the procedure shall be corrected or changed before proceeding to implement the procedure.

Activities that require skills normally possessed by qualified personnel do not require detailed step-by-step delineation in a procedure. They are performed in accordance with documents of a type appropriate to the circumstances such as planning sheets, job descriptions, external manuals, or other form.

## 6.0 DOCUMENT CONTROL

Documents and changes to documents that prescribe or specify quality requirements or activities affecting the availability and/or reliability of IROFS are controlled in a manner that assures the use of correct documents. Such documents, including changes thereto, are reviewed for adequacy and approved for release by authorized personnel.

Procedures and instructions assure that documents are prepared; reviewed for adequacy, correctness, and completeness by a qualified individual; approved for release by authorized personnel; distributed to the location where the activity is performed prior to commencing work; and used in performing the activity. Obsolete or superseded documents are removed or appropriately identified. Procedures identify documents to be controlled; responsibility for

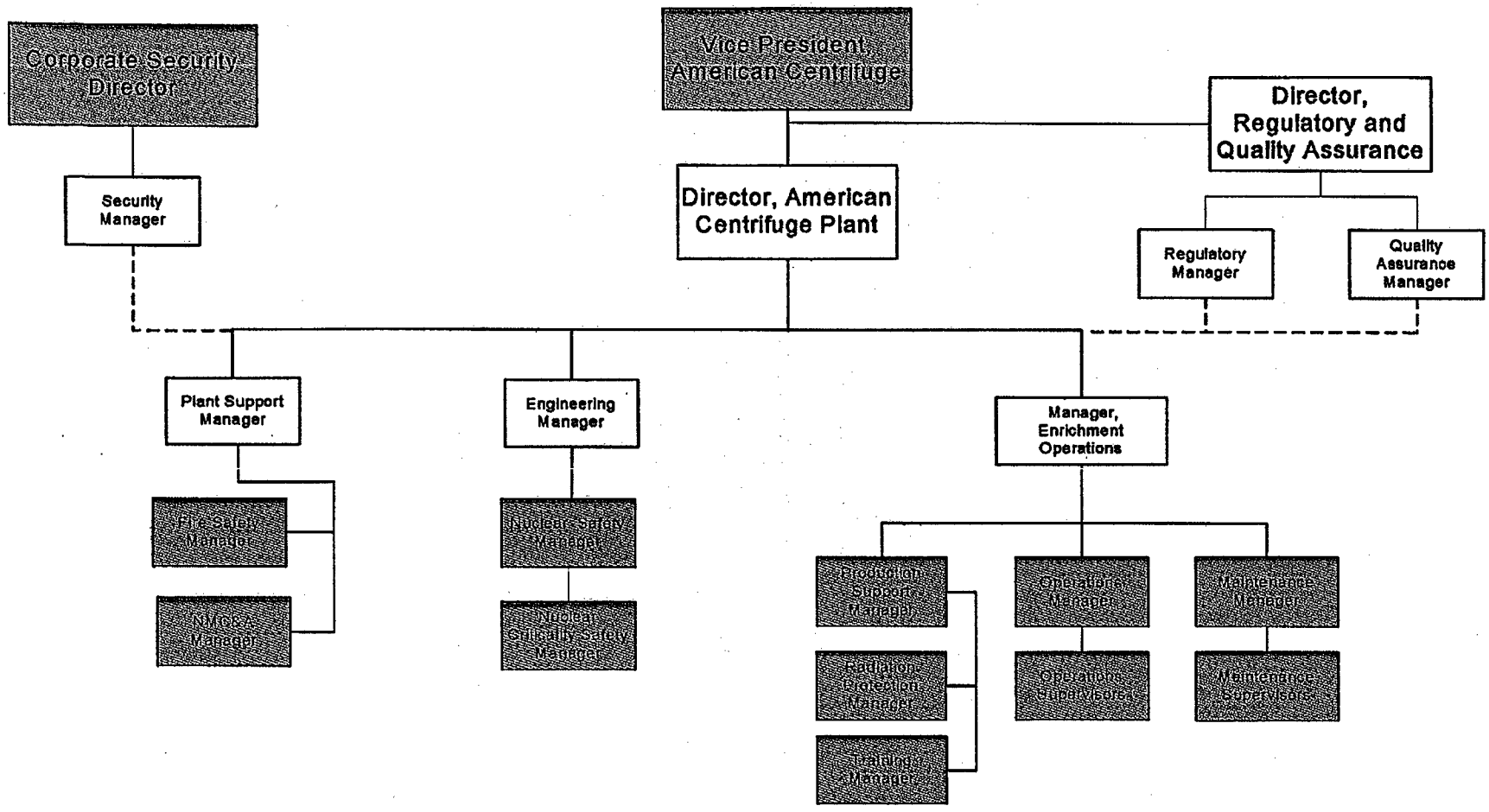


**Figure 1.1-1 American Centrifuge Plant Organizational Chart**

**Information contained within does not contain Export Controlled Information**

**Reviewer: DA Hupp**
**Date: 01/17/06**