LA-2605-0001

License Application

American Centrifuge Lead Cascade Facility

in Piketon, Ohio

Docket No. 70-7003 **Revision 43** **May 2011**

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Reviewer: G. Peed
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LA-2605-0001

LICENSE APPLICATION
FOR THE AMERICAN CENTRIFUGE LEAD CASCADE FACILITY
in Piketon, Ohio

Docket No. 70-7003

Revision 43

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ACRONYMS AND ABBREVIATIONS

ACL	Administrative Control Level
ACP	American Centrifuge Plant
ACR	Area Control Room
AHJ	Authority Having Jurisdiction
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
amsl	above mean sea level
ASME	American Society of Mechanical Engineering
ARA	Airborne Radioactivity Area
BCS	Boundary Control Station
BDC	Baseline Design Criteria
BEQ	Baseline Effluent Quantity
BPCV	backpressure control valve
CA	Contamination Area
CAA	Controlled Access Area
CCZ	Contamination Control Zone
CEDE	Committed Effective Dose Equivalent
CFR	*Code of Federal Regulations*
CIT	Corporate Information Technology
CM	Configuration Management
CTTF	Centrifuge Training and Test Facility
DAA	day accumulation area
DAC	Derived Air Concentration
DAW	dry active waste
DBE	design basis earthquake
DFP	Decommissioning Funding Plan
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EML	Environmental Measurements Laboratory
EP	Emergency Plan
EPA	U.S. Environmental Protection Agency
EPIPs	Emergency Plan Implementing Procedures
ERO	Emergency Response Organization
EV	evacuation vacuum
FCA	Fixed Contamination Area
FDA	Facility Design Authority
FHA	Fire Hazards Analysis
FNAD	Fixed Nuclear Accident Dosimeters
FSRC	Facility Safety Review Committee
GCEP	Gas Centrifuge Enrichment Plant
GDP	gaseous diffusion plant
GET	General Employee Training

HCA	High Contamination Area
HEPA	high efficiency particulate air
HP	Health Physics
HCA	High Contamination Area
HRA	High Radiation Area
HVAC	Heating, Ventilation, and Air Conditioning
ICP/MS	Inductively Coupled Plasma/Mass Spectrometry
IHS	Industrial Hygiene and Safety
IPT	intraplant transporter
IPTT	intraplant tow tractor
IROFS	items relied on for safety
ISA	Integrated Safety Analysis
ISTP	Integrated Systems Test Plan
LCC	local control center
LEC	Liquid Effluent Collector
LLRW	low level radioactive waste
LO/TO	Lockout/Tagout
LSDA	Lower Suspension and Drive Assembly
MAPEP	Mixed Analyte Performance Evaluation Program
MCW	machine cooling water
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentrations
MDU	machine drive unit
MEI	Maximally Exposed Individual
MIP	machine instrument package
MIV	machine isolation valve
MM	Modified Mercalli
MSDS	Material Safety Data Sheet
MSL	mean sea level
M&TE	maintenance and test equipment
NCS	Nuclear Criticality Safety
NCSA	Nuclear Criticality Safety Approval
NCSE	Nuclear Criticality Safety Evaluation
NEPA	National Environmental Protection Act
NESHAP	National Emissions Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NIOSH	National Institute for Occupational Health and Safety
NIST	National Institute of Standards and Technology
NMSZ	New Madrid seismic zone
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
ODNR	Ohio Department of Natural Resources
OJT	on-the-job training
OSHA	Occupational Safety and Health Administration

PGA	peak ground acceleration
PGDP	Paducah Gaseous Diffusion Plant
PB1	X-3001 Process Building
PBT	Performance Based Training
PM	preventive maintenance
PMF	Probably Maximum Flood
PMT	post-maintenance testing
PORTS	Portsmouth Gaseous Diffusion Plant
PPE	personal protective equipment
PSB	Process Support Building
PSM	Process Safety Management
PSS	Plant Shift Superintendent
PTI	permits-to-install
PV	purge vacuum
QA	Quality Assurance
QAPD	Quality Assurance Program Description
QC	Quantity Control
QL	Quality Level
RCRA	*Resource Conservation and Recovery Act* of 1976
RCW	recirculating cooling water
REIRS	Radiation Exposure Information Reporting System
RG	Regulatory Guide
RMA	Radioactive Material Area
RMDC	Records Management and Document Control
RMP	Risk Management Program
RP	Radiation Protection
RPM	Radiation Protection Manager
RQ	Reportable Quantity
RWP	Radiation Work Permit
RGA	Regional Gravel Aquifer
RHW	recirculating heating water
RM	river mile
R/A	Recycle/Assembly
SAR	Safety Analysis Report
SARA	*Superfund Amendments and Reauthorization Act*
SCBA	self-contained breathing apparatus
SME	Subject Matter Expert
SPCC	Spill Protection Control and Countermeasures
STP	Sewage Treatment Plant
SRD	System Requirements Document
SRP	Standard Review Plan
SSCs	structures, systems, and components
TDAG	Training Development and Administrative Guide
TEDE	Total Effective Dose Equivalent
TLDs	Thermoluminescence Dosimeters
TQs	Threshold Quantities

TRM	Training Requirement Matrices
TSD	Treatment, Storage, or Disposal
TWCR	Tower Water Cooling Return
TWCS	Tower Water Cooling Supply
UCNI	Unclassified Controlled Nuclear Information
UCRS	upper continental recharge system
USA	Upper Suspension Assembly
USEC	USEC Inc.
USGS	U.S. Geological Survey
VHRA	Very High Radiation Area

CHEMICALS AND UNITS OF MEASURE

cfs	cubic feet per second
Ci	curie
cm	centimeters
cm^2	square centimeter
dpm	disintegration per minute
F	Fahrenheit
ft	feet
ft/d	feet per day
ft^2	square feet
g	grams
Gal	gallons
Gal/d	gallons per day
HF	hydrogen fluoride
in.	inches
k_{eff}	$k_{effective}$
km	kilometers
km^2	square kilometers
kV	kilovolts
L	liters
lb	pounds
L/d	liters per day
lfpm	linear feet per minute
m	meters
m^2	square meters
mCi	millicuries (one-thousandth of a curie)
mCi/mL	millicuries per milliliter
mg	milligram (one-thousandth of a gram)
mg/L	milligrams per liter
mph	miles per hour
mrem	millirem (one-thousandth of a rem)
pCi	picocurie (one-trillionth of a curie)
pCi/L	piocuries per liter
ppm	parts per million
psi	Pounds per square inch
rem	roentgen equivalent man
SWU	separative work units
UF_6	uranium hexafluoride
wt.	weight
YA	Instrument Air
μCi	microcurie (one-millionth of a curie)
μCi/g	microcuries per gram
μg	microgram (one-millionth of a gram)
μg/kg	micrograms per kilogram
μg/L	micrograms per liter

μg/mL	micrograms per milliliter
μg/m^3	micrograms per cubic meter
μ	micron or micrometer (one-millionth of a meter)

EXECUTIVE SUMMARY

This license application was prepared by USEC Inc. (USEC), for a license to possess and use special nuclear, source, and by-product material in the American Centrifuge Lead Cascade Facility (hereafter referred to as the Lead Cascade) at the Portsmouth Gaseous Diffusion Plant (PORTS) located in Piketon, Ohio under the *Atomic Energy Act* of 1954, as amended, 10 *Code of Federal Regulations* (CFR) Part 70, and other applicable laws and regulations. USEC is the parent company of the United States Enrichment Corporation, which is the current holder of a U. S. Nuclear Regulatory Commission (NRC) Certificate of Compliance for PORTS issued under 10 CFR Part 76. USEC is a global energy company and the world's leading supplier of enriched uranium fuel.

The Lead Cascade is a test and demonstration facility designed to provide information on American Centrifuge enrichment technology to support the development, construction, and operation of the American Centrifuge Plant (ACP). The Lead Cascade is an important step toward advancing the national energy security goals of maintaining a reliable and secure domestic source of enriched uranium. These goals are consistent with the purposes for which the Corporation was created. Through amendments to the *Atomic Energy Act*, Congress created the Corporation to, among other things, conduct research and development, as required, to evaluate alternative technologies for uranium enrichment, and to help maintain a reliable and economical domestic source of enriched uranium.

The Licensee is responsible for the design, fabrication, installation, operation, maintenance, modification, and testing of the Lead Cascade. The goal of the project is to provide reliability, performance, cost and other data vital to making decisions concerning the deployment of a Commercial Plant and to reduce the financial risk of such deployment. The Lead Cascade operates up to 240 centrifuge machines in the recycle mode as "closed loop" systems, where the enriched product stream is recombined with the depleted stream prior to being re-fed to the cascade. The Lead Cascade uses full-scale equipment and laboratory samples are withdrawn to confirm the enrichment process. It is operated so that no enriched material is withdrawn, other than laboratory samples. The Lead Cascade may possess up to 250 kilograms uranium hexafluoride and may enrich uranium up to 10 weight percent ^{235}U. The design of the Lead Cascade complies with the Baseline Design Criteria specified in 10 CFR 70.64(a) and the defense-in-depth requirements contained in 10 CFR 70.64(b).

The Lead Cascade is located on land leased from the U.S. Department of Energy (DOE)- in rural Pike County, a sparsely populated area in south central Ohio. The PORTS reservation has been studied and characterized extensively by both DOE and USEC. The facilities utilized for the Lead Cascade, which are part of the former DOE Gas Centrifuge Enrichment Plant program, were built in the early 1980s. No new facilities will be constructed because the infrastructure needed to operate a Lead Cascade is already in place. The existing facilities will be refurbished to accommodate the Lead Cascade. In addition, the Lead Cascade will use other existing site-wide services such as laboratory analysis, fire protection, security, medical, emergency management, waste management, and environmental monitoring.

This license application follows the format and guidelines provided in NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*. The Application is written prospectively in the present tense, representing the licensed condition. The information provided reflects the design in sufficient detail to enable a reviewer to make a definitive evaluation that the Lead Cascade can be constructed and operated without undue risk to the health and safety of the public and with no significant impact to the environment.

1.0 GENERAL INFORMATION

The American Centrifuge Lead Cascade Project (hereafter referred to as the Lead Cascade) encompasses installation and operation of up to 240 gas centrifuge machines within existing buildings, located on the site of the Portsmouth Gaseous Diffusion Plant (PORTS). PORTS is operated by USEC's wholly owned subsidiary, the United States Enrichment Corporation (Corporation), under a Certificate of Compliance issued by the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 *Code of Federal Regulations* (CFR) Part 76. The goal of the project is to provide reliability, performance, cost and other data vital to making decisions concerning the deployment of a Commercial Plant and to reduce the financial risk of such deployment. The facility may enrich uranium up to 10 weight (wt.) percent ^{235}U. The cascade is operated on recycle where the enriched product stream is recombined with the depleted stream prior to being re-fed to the cascade. Samples are taken for laboratory analysis. The Licensee's possession limit of uranium hexafluoride (UF_6) for the Lead Cascade is 250 kilograms (kg). The Lead Cascade's license initially authorizes operation for a period of five years. The expiration date is specified in Materials License SNM-7003.

This license application follows the format and guidelines provided in NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*. The Application is written prospectively in the present tense, representing the licensed condition. The information provided reflects the design in sufficient detail to enable a reviewer to make a definitive evaluation that the facility can be constructed and operated without undue risk to the health and safety of the public and with no significant impact to the environment.

1.1 Facility and Process Description

The Lead Cascade operates up to 240 centrifuge machines in the recycle mode as "closed loop" systems, where the enriched product stream is recombined with the depleted stream prior to being refed to the cascade. Additional centrifuges may be available for other uses, but are not installed for operation (e.g., spares). The Lead Cascade uses full-scale equipment and laboratory samples are withdrawn to obtain information on American Centrifuge enrichment technology. The Lead Cascade is operated so that no enriched material is withdrawn, other than laboratory samples. The Lead Cascade may consist of multiple individual cascades that operate independently. An individual cascade may be in operation while other cascades are being installed or removed. No finished product is produced by the Lead Cascade.

1.1.1 Facility Layout

The Lead Cascade is located within the Licensee-leased area of the U.S. Department of Energy's (DOE) PORTS reservation depicted in Figure 1.1-1 (located in Appendix B of this license application). The facilities utilized for the Lead Cascade, which are part of the former Gas Centrifuge Enrichment Plant (GCEP) program, were built in the early 1980s. Construction of the ACP has begun so the GCEP facilities are now known as the ACP.

The design of the Lead Cascade complies with the Baseline Design Criteria (BDC) specified in 10 CFR 70.64(a) and the defense-in-depth requirements contained in 10 CFR 70.64(b).

The Lead Cascade facilities shown in Figure 1.1-2 (located in Appendix B of this license application) include the X-3001 Process Building (PB1), which houses up to 240 operating centrifuge machines, associated process piping, instrumentation and controls, computer systems, and auxiliary support equipment. The facilities also include the X-3012 Process Support Building (PSB) to provide oversight and control of the equipment in the cascade. The X-7726 Centrifuge Training and Test Facility (CTTF) provides areas to receive and test centrifuge components, and to assemble and repair the centrifuges. A transporter moves centrifuge machines between the CTTF and PB1 through the covered X-7727H Transfer Corridor. The X-3012 also provides offices, lockers, change rooms, and break rooms. A portion of the X-7725 Recycle/Assembly (R/A) Building provides similar administrative facilities, buffer storage area for storage, handling, and assembly of centrifuge components and storage and handling of completed centrifuge machines, as well as training rooms, and the storage and maintenance areas for the transporter. PORTS facilities that provide support to the Lead Cascade Facility include XT-847 Waste Management Staging Facility and X-710 Technical Services Building.

In addition to these facilities, the Corporation also provides support facilities for the Lead Cascade. Facilities for emergency response, training, maintenance, laboratory support, utilities, environmental and waste management support, and administrative support are provided through existing facilities at PORTS. A brief description of the Lead Cascade facilities and their functions is provided below.

1.1.1.1 X-7726 Centrifuge Training and Test Facility

The CTTF contains approximately 30,000 square feet (ft^2) of floor space at ground level located in the northwest corner of the R/A Building. The CTTF is the area where material and components are received; components or subassemblies are inspected and tested; the components are assembled as centrifuge machines; the final assembly is evacuated and leak checked; and repairs are performed to the machine or subassemblies. A drawing depicting the assembly stand is provided in Figure 1.1-3 (located in Appendix B of this license application). The functional areas of the CTTF include the following:

- Machine assembly stands/repair stands
- Subassembly diagnostic/test stands
- Receipt of subassemblies, components, and materials
- Storage for centrifuges, casings, rotors, and columns
- Inspection, testing, and repair of large subassemblies
- Storage for small components and subassemblies
- Inspection and testing of components
- Assembly of subassemblies
- Functional testing of subassemblies

- Auxiliary systems area
- Electrical substation
- Backup generator

An overhead crane system traverses the length of the CTTF for movement of centrifuge machines or other large components.

1.1.1.2 X-7725 Recycle/Assembly Building

The X-7725 is a very large multiple level building. Floor areas of the R/A Building are shared with ACP. The Lead Cascade portions of the R/A Building consist of a small area along the west portion of the building to allow storage, handling and assembly activities of centrifuge components, and either temporary storage or movement of completed centrifuge machines by overhead crane from the adjoining X-7726 to a ground level transporter for storage or final movement to the Lead Cascade. This area also includes two rooms available to support various maintenance activities and to support manufacturing and assembly activities. Administrative offices, lockers, change rooms, break rooms, and training rooms are also available in the X-7725. An area in the X-7725 R/A 3rd level storage area between column lines C3 and C8 will be used for centrifuge component handling and storage. Areas of the X-7725 will be utilized for shipping, receiving and storage of Lead Cascade related materials.

1.1.1.3 X-7727H Transfer Corridor

The X-7727H Transfer Corridor provides an enclosed throughway from the X-7725 to the PB1. The corridor is an approximate 30 feet (ft) wide by 800 ft long aisle. The space is environmentally controlled and has floor to ceiling, power operated, double doors at each end to enable temperature control between facilities. Transfer corridor door actuators operate the doors between the Transfer Corridor and PB1, between the X-3002 Process Building and the PSB, and between the Transfer Corridor and the R/A Building. Two sets of doors form a barrier between the Transfer Corridor and the R/A Building to maintain the conditioned environmental controls of the R/A Building.

The surface of the transfer corridor is of concrete construction with a smooth troweled finish and is wide enough and of sufficient strength to permit two loaded transporters to pass. It is at right angles to the centrifuge aisles in PB1 and extends from the Lead Cascade process area to the CTTF. The corridor surface has expansion joints, which are perpendicular to the path of travel and are keyed together with steel to inhibit differential settlement.

1.1.1.4 X-3001 Process Building

PB1 is a high bay building housing the Lead Cascade consisting of up to 240 operating centrifuge machines in multiple cascades that operate independently. The transporter delivers assembled centrifuge machines to PB1 where a rigid mast crane removes each centrifuge machine and installs it into a cascade position. The centrifuge machine is connected to a service module position where the centrifuge is supplied with auxiliary utilities, power, controls, and UF_6. The Lead Cascade is supplied normal (approximately 0.711 wt. percent ^{235}U) UF_6 from

either a Model 5A/B, 8A, or 12B UF_6 cylinder through a feed system consisting of a portable cart capable of heating the solid material to a gaseous state. After the initial fill from the portable feed cart, the centrifuge machines operate on a recycle mode as a “closed loop” system in the gaseous state, and the feed cart is on “standby.” This recycle mode in the “closed loop” causes the enriched material within the cascade to be mixed with the depleted material within the cascade prior to it re-entering the feed stage. Laboratory quantities of UF_6 are sampled from the Lead Cascade in order to perform analyses. No enriched product is withdrawn from the cascade, except for the samples. Samples are processed in the X-710 Analytical Laboratory. The cascade enrichment is normally less than 5 wt. percent assay. However, testing of the cascade may result in some material being enriched above 5 wt. percent, with a licensed limit of 10 wt. percent ^{235}U.

A dump cart is provided to remove the contents of the cascade in the event inventory must be reduced for normal operations or as a result of upset conditions. A local control center (LCC) at the cascade provides operator interface through controls and instruments with the centrifuge machines, and an area control room (ACR) located in the X-3012 also provides limited control of the centrifuges remotely.

The north end of PB1 has an equipment/utility mezzanine where auxiliary equipment is housed. Items in this area consist of heating and ventilating equipment, cooling water pumps, vacuum pumps, and electrical switchgear. A building vent for the purge and evacuation vacuum systems is also located in PB1. Due to the nature of centrifuge operation, a purge vacuum is applied to the machine to remove any gas (either process gas or in-leakage of atmospheric gases) that enters the space between the internal rotor and the casing. The vent is monitored and is permitted through the Ohio Environmental Protection Agency (EPA).

1.1.1.5 X-3012 Process Support Building

The PSB is located east of PB1. The PSB houses the ACR, maintenance shops and stores, offices, men’s and women’s lockers and restrooms, and a lunchroom. A high bay (60 ft clear height) transfer corridor divides the PSB between the operations and maintenance functions.

1.1.1.6 Support Facilities

In addition to the primary facilities described above, several other PORTS facilities provide support and/or services for the Lead Cascade. Utilities provided to the Lead Cascade include electrical, sanitary water, sanitary sewer, instrument air, communications, and non-potable cooling water. Also, emergency response, training, maintenance, laboratory, environmental management, security, and administrative support are provided through existing facilities and services at PORTS. Two of the principal support facilities are discussed below.

1.1.1.6.1 XT-847 Waste Management Staging Facility

The XT-847 Waste Management Staging Facility is located near the southern end of the PORTS reservation. The building is a steel structure with concrete floors and is divided into three major staging areas. The northern and southern sections are separated from the center

section of the building by concrete block four-hour rated firewalls and steel fire doors. An administrative area adjoins the staging area.

The facility is used to accumulate and stage/prepare radioactive waste and non-hazardous recyclable materials prior to shipment off-site. The building is equipped with truck and rail loading/unloading facilities and scales.

1.1.1.6.2 X-710 Technical Services Building

The X-710 Technical Services Building Analytical Laboratory performs chemical and isotopic analyses for the Lead Cascade.

Included in these services are sample specification analyses, enrichment performance analyses, Nuclear Criticality Safety limitation analyses, uranium accountability analyses, and mass spectrometry isotopic analysis; vent emission analyses; and High Efficiency Particulate Air filter testing. The laboratory routinely performs inorganic, organic, asbestos and radiochemistry analyses and physical properties measurements of samples in support of various programs and facility operations. Further, the laboratory provides technical support for non-destructive analysis of equipment and materials.

1.1.2 Process Description

This process description is broken into five sections that describe the primary gas centrifuge process: 1) general separation fundamentals, 2) centrifuge separation fundamentals, cascade theory, 4) design of the gas centrifuge, and 5) operation of the gas centrifuge.

Other operations that are performed to support the primary process include equipment and machinery repair and fabrication of specialized equipment. These activities may be conducted with equipment contaminated with uranium-bearing material. The uranium-bearing material could be UF_6, uranium tetrafluoride (UF_4), uranyl fluoride (UO_2F_2), or an intermediate oxy-fluoride.

1.1.2.1 General Separation Fundamentals

The processing of UF_6 into an isotopic content that enables most commercial nuclear reactors to produce electricity through a controlled fission reaction is called enrichment. The enrichment process increases the concentration of the fissionable ^{235}U isotope from its naturally occurring assay of approximately 0.711 wt. percent to a preferred range of 2 to 5 wt. percent assay. The balance of uranium consists primarily of the ^{238}U isotope.

There are two methodologies of enrichment commercially employed, the gaseous diffusion process and the gas centrifuge process. Both processes consist of the interconnection of multiple “separation elements” in configurations known as a cascade. Figure 1.1-4 is a diagram of a separation element, consisting of a feed stream (F) that is separated into product (P) and tails (T) streams. The concentrations of ^{235}U in the feed, product and tails streams are N_F, N_P and N_T, respectively.

The amount of effort required to increase (enrich) a given quantity of uranium from concentration N_F to concentration N_P is described in terms of separative work units (SWU). Separative work is a descriptive mathematical quantity that encompasses the separation factor, ability of a separation element to separate ^{235}U and ^{238}U, and the material flow of a separation element.

1.1.2.2 Centrifuge Separation Fundamentals

Figure 1.1-5 shows a simplified schematic of a gas centrifuge machine. A centrifuge machine consists of a large rotating cylinder and piping for the feeding of UF_6 gas and the withdrawal of depleted and enriched UF_6 gas streams. The rotating cylinder, called the rotor, is contained within another cylinder, called the casing, which maintains the rotor in a vacuum and provides physical containment of components in the unlikely event of a catastrophic failure of the machine. Other major components of a centrifuge include upper and lower suspension systems and a motor and control system.

For an operating centrifuge, UF_6 gas is fed into the rotor, which is spinning at relatively high rotational velocities. The heavier $^{238}UF_6$ isotope accumulates at the rotor wall, whereas the lighter $^{235}UF_6$ isotope accumulates more toward the center (pushed away from the wall by the $^{238}UF_6$ isotopes). A slight axial (counter current) flow, induced by mechanical and/or thermal agitation, carries the $^{238}UF_6$ downward along the wall and the $^{235}UF_6$ upward along the axis. As the gas travels up the axis of the centrifuge, it is constantly being depleted in $^{238}UF_6$ and enriched in $^{235}UF_6$. So the longer the centrifuge and the faster gas can be transported from one end of the centrifuge to the other, the higher the centrifuge machine's separative capacity will be. The combined effect of the radial gradient and axial flow enables a relatively significant assay gradient to develop between the bottom and the top of the centrifuge.

The separation capacity of a centrifuge is the function of two phenomena: one, the radial separation, and two, the axial separation. Radial separation (separation factor) is created by centrifugal force. Axial separation is created by the net transport of $^{235}UF_6$ to the top and $^{238}UF_6$ to the bottom of the centrifuge. The separation factor of the centrifuge process is an order of magnitude higher than that of the gaseous diffusion process, although neither is much greater than a factor of one. Due to the higher separation factor of the centrifuge process, there are also orders of magnitude fewer stages required in a centrifuge facility than in a gaseous diffusion plant (GDP).

1.1.2.3 Cascade Theory

Separating elements are connected in series, called stages, to achieve the desired assay of ^{235}U. Many separating elements are also connected in parallel in the centrifuge process to achieve the desired mass flows, forming a cascade. Figure 1.1-6 shows a schematic of a typical cascade that takes on a "diamond" shape. Natural feed enters the cascade at the middle of the diamond, with product streams being enriched in ^{235}U to the top of the diamond and the tails streams being depleted of ^{235}U to the bottom of the diamond. There are nine stages in the example cascade shown, including a feed stage, five enrichment stages and three stripping (depletion) stages.

1.1.2.4 Design of the Gas Centrifuge

The gas centrifuge machine is comprised of a number of subassemblies (see Figure 1.1-5):

- Casing
- Rotor
- Column
- Upper Suspension Assembly (USA)
- Lower Suspension and Drive Assembly (LSDA)
- Diffusion Pump (not depicted in figure)

Degradation or failure of any of these key components can have a significant impact on the operation of the centrifuge machine. The most significant failures that can occur involve the rotor, and the upper and lower suspension assemblies. Complete failure of any of these components that cause a failure of the rotor is called a “crash” and requires removal, replacement, or isolation of the entire machine. On the other hand, failure of the other components, such as the thermal shield, is more likely to cause degradation in the performance of a machine, but is not expected to lead to a machine crash.

1.1.2.4.1 Casing

The casing is the outermost cylinder of a centrifuge and serves several functions. Foremost, the casing acts as a means of containment of rotor debris in the unlikely event of an internal failure, protecting personnel and adjacent machines from harm. The casing is also a vacuum chamber for the rotor. It is important to maintain the outside of the rotor in vacuum because any gas that impacts the rotor’s exterior wall causes drag, which in turn causes localized heating of the rotor and consumes power from the motor, thereby slowing it down. The casing also acts as a structural support for the rotor, the upper suspension, and column assemblies.

1.1.2.4.2 Rotor

The rotor in a centrifuge consists of a thin walled rotating shell with end caps on both ends. The rotor is designed to contain process gas while spinning at relatively high peripheral velocities enabling the enrichment of the desired isotope. A number of factors are taken into account when designing a rotor such as materials of construction and speed of operation. Rotors are often made of light, high strength-to-weight ratio materials, such as carbon fiber reinforced plastics.

In choosing the material of construction for the rotor, several characteristics are of importance:

- Ultimate tensile stress
- Modulus of elasticity
- Density
- Resistance to corrosive attack by UF_6

The ultimate tensile strength of the material determines the maximum peripheral velocity a rotor can survive before bursting. As the ultimate tensile strength increases and the density decreases, the maximum operational velocity of the rotor and therefore, SWU output increases.

The rotor geometry, length and radius, and the material of construction's modulus of elasticity and density determine the critical frequencies of the rotor. A critical frequency occurs when the natural resonating frequency and the rotational frequency of the rotor coincide. When a critical frequency is reached the rotor starts to vibrate like a plucked guitar string. These vibrations cause off-normal stresses in the suspension system that can lead to failure.

A rotor can have multiple critical frequencies, with each increasing frequency resulting in a greater number of nodes. In general, a centrifuge is called a subcritical or a supercritical machine, if it is operating below or above its first critical frequency, respectively.

The rotor material's ability to resist corrosion is crucial to the operational life of a centrifuge. A rotor that does not have corrosion resistance to the process gas will experience degradation of the very mechanical properties that allow it to operate.

1.1.2.4.3 Column

The process piping internal to the centrifuge machine is called the column. The column provides a means to introduce UF_6 gas (feed) into the machine while removing enriched UF_6 gas (product) and depleted UF_6 gas (tails). When designing the column, care is taken to keep the structure as rigid as possible, thereby maintaining machine alignment and tolerances within operating limits. As a result of the opening created by the column passing into the rotor, some lighter gases are free to enter the space between the rotor and the casing where it is removed by the diffusion pump.

1.1.2.4.4 Upper Suspension Assembly

The USA consists of a magnet that carries a fixed percentage of the rotor weight and is designed to compensate for changes in the length of the rotor as it is driven to speed.

1.1.2.4.5 Lower Suspension and Drive Assembly

The LSDAs use a motor and a suspension system to drive and support the rotor, respectively. The management of heat within the lower suspension and drive assembly is important to maintaining a controlled thermal gradient within the rotor.

1.1.2.4.6 Diffusion Pump

The diffusion pump maintains the casing vacuum by continually removing any gas, either process gas or in-leakage of atmospheric gasses, that enters the space between the rotor and the casing. As previously described, it is important to keep gas molecules from impacting the outside of the rotor.

1.1.2.5 Operation of the Gas Centrifuge

The arrangements of the centrifuges in the UF_6 enrichment process are selected to minimize the likelihood of a major interruption of operations. This design concept favors the use of small, separate, systems such that individual centrifuges or entire cascades can be isolated to minimize losses due to abnormal operating circumstances. A primary purpose of isolation is to prevent or limit the transport of light gases to centrifuges that are operating satisfactorily. Light gases can be introduced from leaks, misoperation of the UF_6 feed system, and centrifuges that are encountering operational problems.

1.1.2.5.1 Service Module

Within the process building, utilities and process piping are routed to the centrifuge machines via service modules that consist of a welded, square-tube steel frame structure with pipe headers and valves, control and instrument cabling, ventilation ductwork, and electrical distribution cables running the full length. Pipe headers for process gas and vacuum are aluminum or stainless steel, while those for air, cooling water, and fire suppression are steel. Smaller branch pipes connect the headers to each of the centrifuge machines. The machine isolation valves (MIVs), machine power controls, and machine instrumentation are also mounted on the service modules. Each service module services twenty centrifuge machines and the service modules are connected in series to support an operating cascade.

1.1.2.5.2 Intermachine Flow and Control

The intermachine flow and control system consists of:

- The process piping headers and valves for transporting the process gas;
- The feed control system for controlling the cascade feed rate;
- The inventory control system for maintaining the proper tails backpressure on each machine;

- Instrumentation and controls for header pressures and centrifuge machine status; and
- The sampling system for measuring product and tails assays and product contaminants.

1.1.2.5.3 Process Piping Headers and Valves

Centrifuges for the cascade are arranged in a series of stages. Figure 1.1-7 shows a multi-stage configuration and the flow arrangement between stages. Isolation valves for the individual centrifuge are contained in a MIV set that takes valve mode commands from a machine instrument package (MIP). The Lead Cascade is operated in the cascade recycle mode and the feed/total cascade recycle header receives flows from the cascade tails (depleted material) and product (enriched material) headers. The source for the initial feed is a feed cart and withdrawal from the tails or product headers is performed with a sample cart for the purpose of obtaining sample quantities for laboratory analysis. Figure 1.1-8 (located in Appendix B of this license application) depicts the Lead Cascade system interfaces.

1.1.2.5.4 Feed Control System and Dump System

The feed system for the Lead Cascade is a portable cart (Figure 1.1-9) designed to heat, if required, a cylinder of solid normal UF_6 to vaporize the solid for feeding. Prior to feeding UF_6 from the feed cylinder to the centrifuge machines, the cylinder may be burped to remove contaminants (lights) that would impact the operation of the machine. This is accomplished utilizing the dump cart (Figure 1.1-10). The UF_6 is fed to the cascade, and the Lead Cascade is operated in the recycle mode after the initial fill. The feed cart is placed on standby. In the event of an operational upset or other condition requiring removal of inventory, a dump cart is connected to the cascade during recycle operations to implement removal of the UF_6 from the cascade. The dump cart cylinder may contain depleted UF_6.

1.1.2.5.5 Inventory Control System

Regulating the tails header pressure in each stage is the typical means to control cascade inventory. This method permits control of the tails flow from each stage and minimizes inventory changes in the centrifuges. The inventory control system consists of a backpressure control valve (BPCV) and a reference pressure (datum) line to remotely adjust the setpoint of the BPCV. Datum pressure controls and monitoring capabilities of the control valve position are provided at the LCC and the ACR.

This portion of the text has been determined to contain Export Controlled Information and is located in Appendix B of this license application.

1.1.2.5.6 Instrumentation and Controls

The LCC or the ACR provide the operator interface with controls, status indicators, alarms, and digital displays. Cascade feed, product, and tails pressures and temperatures are monitored, displayed, and alarmed at the LCC or the ACR. Additional sensors/transmitters are located at different positions on the outside casing of several specific centrifuges and along the service module to provide additional information for test purposes. Some centrifuge casings are

equipped with interface connectors for monitoring other machine parameters. Valve control commands are provided and the valve positions are monitored and alarmed at the LCC or the ACR. Centrifuge isolate/de-isolate and dump commands, speed controls, and centrifuge status are provided remotely at the LCC or the ACR. Automatic controls and programmed software protect equipment in the event of abnormal operations.

1.1.2.5.7 Sampling System

A sample cart (Figure 1.1-11) is provided to withdraw process gas samples for laboratory analysis to confirm the enrichment process. No enriched material is withdrawn, other than laboratory samples. In addition, contaminants in form of light gases (e.g., oxygen, nitrogen, and carbon dioxide) are monitored by taking grab samples from sample taps on the product and tails headers of each stage.

1.1.2.6 Machine Assembly

The centrifuge machines for the Lead Cascade are assembled in the CTTF. Parts for Lead Cascade centrifuge machine assembly are received at this location. Secure facilities are available to receive and store the classified parts as well as other components of the centrifuge machines. Additional parts receiving and storage space is available if necessary in the adjacent R/A Building. Overhead cranes, fork trucks, and a parts elevator are available to handle timely parts delivery to the CTTF assembly stand for assembly. Some assembly activities may occur in the R/A Building and PB1. Some centrifuge parts are surface coated for protection to ensure longer life cycles. Curing capability may be necessary to ensure parts are available in a timely fashion. Electric heaters can be used to support the curing process.

Two machine assembly positions and a column assembly stand are provided in the CTTF assembly stand for assembly of the various components into a completed machine. Overhead cranes are available for material handling needs including long parts insertion and lower and upper assembly installation. Lifting fixtures and other assembly tooling are also required during the assembly of the Lead Cascade machines. Gross leak/vacuum testing is performed at this location before the assembled machine is moved from the CTTF assembly stand. No process gas (UF_6) testing of the machines take place in the CTTF. Completed machines may be moved via crane to an adjacent storage location until they can be moved, again by crane or moved directly, to a transporter for movement to the process building. Testing of the machines using UF_6 is performed in PB1 after installation, prior to being placed into service.

1.1.2.7 Lead Cascade Support Systems

After assembly of the machine, several additional systems are provided to move the completed machine to its proper location and for connection and checkout prior to start-up.

1.1.2.7.1 Purge Vacuum/Evacuation Vacuum

The high peripheral velocity of a gas centrifuge requires the rotor to operate in a high vacuum to minimize friction. Each centrifuge casing is therefore fitted with a diffusion pump to produce the required vacuum between the rotor and the casing. The purge vacuum (PV) system for the Lead Cascade maintains a suitably low pressure for efficient operation of the diffusion pumps during normal operation. The output of the diffusion pumps discharges to the PV system. Any UF_6 and light gases that may escape from the rotor and any light gases entering the vacuum system due to in-leakage are removed.

Each PV system includes two mechanical vacuum pumps each equipped with automatic valves, power, and controls to enable one vacuum pump to serve as a spare for the other. The PV system also includes the piping network necessary to accommodate the flow of the effluent gases as they pass from the centrifuge diffusion pump to the atmospheric vent. Four chemical traps, and an exhaust gas analyzer with UF_6 and gas flow monitoring capability are provided as part of the system. One or more PV systems are utilized depending on the number of installed machines.

The main sources of gases to be removed from the centrifuge by the PV system are:

- Air leakage into the casing, process lines, or PV system;
- Hydrogen fluoride that originates from the cascade feed and from the reaction of UF_6 and moisture from air in-leakage;
- UF_6 leakage into the centrifuge casing vacuum; and
- Residual inert gas from UF_6 feed cylinders.

This portion of the text has been determined to contain Export Controlled Information and is located in Appendix B of this license application.

The EV system includes two mechanical vacuum pumps, valves and controls to permit one vacuum pump to serve as a spare for the other. The system also includes the piping network required to connect the centrifuges from the diffusion pump through the service module piping to the mechanical vacuum pumps, and piping from the discharge of the mechanical vacuum pumps to the chemical traps that are also part of the PV system.

Gases vented by the PV and EV systems are monitored to ensure proper operation of chemical traps to minimize any potential release of radionuclides. The EV system has the capability to bypass the chemical traps during initial start-up and to pump down service modules, piping, and new machines prior to gas introduction.

1.1.2.7.2 Machine Cooling Water System

The machine cooling water (MCW) system, shown in Figure 1.1-13, is a closed-loop circulating water system designed to provide cooling of the centrifuge diffusion pump, LSDA, and the PV and EV vacuum pumps.

The MCW system serves the Lead Cascade and associated mechanical vacuum pumps. This system contains circulating water pumps, filter, heat exchanger, an expansion tank, and a piping tie-in to the chemical feed, deionizer, and sanitary water systems.

Heated MCW leaves the centrifuge cascade through the service module header to an expansion tank, which provides enough suction head for the MCW circulating pumps. The tank provides a convenient point for adding make-up water and the water treatment chemicals. The discharge of the circulating pumps passes through a MCW filter to a heat exchanger. The heat exchanger cooling water (TWC) is supplied from the cooling tower, where the TWC is cooled. The cooled MCW then returns to the centrifuge machines by way of the supply header in the service module.

The MCW system requires a chemical feed system where water treatment chemicals are added. The chemical feed system contains a chemical tank from which chemicals are added via a chemical injection pump.

Sanitary water is provided for MCW makeup water. This water passes through a deionizer before entering the MCW closed loop. The makeup water is used for initial fill purposes and for maintaining the proper level of MCW in the system. MCW system alarms are monitored in the ACR. In the event that the Plant Cooling Tower is unavailable, sanitary water may be used as emergency backup to MCW system with the return water being discharged to the sanitary sewer system.

1.1.2.7.3 Machine Mount System

The machine mount system is the primary structural interface between the soil subgrade of the PB1 floor and the centrifuge machines. The machine mount system is a hard-torsion, hard-shear, and soft-rocking system consisting of two groups. The floor module group contains the frame and the reinforced concrete. The machine-mounting group contains the isolation subsystem and the support subsystem.

The mount system is designed so that each machine responds to its operating environment independently of all other machines. This is accomplished by having a stiff, massive concrete and steel-reinforced mat that mitigates the effects of torque and shear forces experienced during rotor failures (operational upset). The overturning forces experienced during rotor failure are mitigated by the machine mount system's soft-rocking suspension. This soft-rocking suspension system also attenuates the vibration response from external excitation such as an earthquake. Elastomeric isolators are arranged in a symmetrical pattern about the fifth point

support to provide a uniform response to a possible random direction of applied ground motion loads.

1.1.2.7.4 Machine Transport System

The machine transport system consists of (1) the X-7726 CTTF Bridge Crane and Monorail system, (2) the X-7725 R/A buffer storage area crane system, (3) the Centrifuge Machine Transporter, (4) the X-3001 Process Building RMC, and (5) the Crane Rails. This system provides a means to transport machines between the assembly area in CTTF and the process building, and install and remove centrifuge machines in the LC operating positions.

The X-7726 CTTF crane and monorail system is used to move centrifuges between the CTTF assembly area and the X-7725 buffer storage area. This system may also be used to load and unload the transporter.

The X-7725 R/A buffer storage area crane is used move centrifuges within the buffer storage area floor stands, and to load and unload the transporter.

The transporter is a battery-operated piece of mobile equipment designed and built to safely transport one or more centrifuges in a vertical position while protecting the machine from excessive motion. The transporter is used to move centrifuges between the CTTF assembly area (and the adjacent R/A buffer storage area) and the process building via the X-7727H Transfer Corridor.

The rigid mast crane consists of a 7 ½-ton capacity, electric-powered, top-running, double-girder, overhead bridge that carries a top-running trolley. The trolley incorporates a rotating platform, which is mounted on a large-diameter bearing near the center of the trolley to provide 370^{o} rotation. The rotating platform supports hoist machinery, a vertical rigid mast, and the operating cab. The hoist raises and lowers a load carriage, which travels along mechanical guides on the rigid mast. The rigid mast serves to prevent the load from swinging during movement of the rigid mast crane. An upper load-lifting yoke and a load-stabilizing clamp are provided on the load carriage to lift and hold a centrifuge. The rigid mast crane is controlled in both the manual and programmed positioning mode by the operator in the crane cab, or during precise positioning operations, by an operator at floor level using a hand-held, reel-mounted pendant control. The pendant control contains a direct communication line between the cab operator and the floor operator.

1.1.2.7.5 Heating, Ventilation, and Air Conditioning

The Lead Cascade heating and ventilation systems are designed to maintain the environment required for proper operation of the process and associated equipment in the process buildings. The main subsystems affecting X-3001 are the Process Area Ventilation System and the Process Area Heating and Pressurization System.

The Process Area Ventilation System provides circulation of air and maintains a positive pressure with respect to the outside ambient atmospheric pressure in X-3001 to reduce the infiltration of dirty and/or cold air. Each ventilation unit consists of a supply fan, a return/exhaust fan, filters, and associated ductwork with automatic dampers, and controls. The return/exhaust air fan draws heated air from the centrifuge machine area and, depending on the building temperature, exhausts it to the outside or recirculates it to the supply fan plenum. If it is necessary to cool the process building, outside air is drawn through a damper into the supply fan plenum where it mixes with air from the return/exhaust fan and passes through a filter to the supply fan inlet. The supply fan discharges through a damper into a large duct located along the

length of the cascade on top of the service module piping. Tempered air is directed from the service module duct outward to the centrifuges. No heating coils are provided in this system. The Process Area Ventilation System can also be used for smoke removal in the event of a fire in the X-3001.

The Process Area Heating and Pressurization System heats the outside make-up air to building temperature and supplies enough heat to offset exterior wall and roof heat losses. Individual heating and pressurization units are located on the mezzanine in X-3001. Each unit consists of a pneumatically operated outside air intake damper, a return air damper, a filter section, a heating coil section, a supply fan, and distribution ducts. The outside air and return air dampers are modulated to maintain a positive building pressure. Recirculating Heating Water (RHW) supplied from the DOE RHW Boiler System is supplied to the heating coils.

Heating, ventilation, and air conditioning (HVAC) are provided to the X-3012 to provide proper operation of the equipment, as well as comfortable working conditions for Lead Cascade personnel. Other areas of the Lead Cascade are provided with HVAC or only heating and ventilation, depending on location and function of the area. Supplemental heat can be provided in the Lead Cascade Facilities using portable electric heaters should the RHW be out of service or outside weather conditions dictate the need.

1.1.2.8 Plant Support Systems

In addition to the Lead Cascade support systems described above, the two primary utility support systems provided through existing GDP support systems are electrical supply and instrument air.

1.1.2.8.1 Main and Auxiliary Power System

The Lead Cascade obtains its electrical power from the same complex of incoming high-voltage transmission lines as the GDP via the X-530 Switchyard, but utilizes a separate set of step-down transformers to supply all the electrical power requirements for the Lead Cascade facilities via 15 kilovolt (kV) feeders from the X-5001 Substation. Each power transformer is sized to carry the complete load of the respective substation. The switchgear is arranged so that all feeders can be fed from any transformer in the respective substation.

The Lead Cascade 15 kV power distribution system shown in Figure 1.1-14 is a “radial” system (i.e., no paralleled source of power). Essential loads in the Lead Cascade are supplied from double-ended substations in which a bus-tie breaker can be closed when power is lost to one end of the substation and portions of which are further supported with standby power such as engine-generators.

1.1.2.8.2 Instrument Air System

The Instrument Air (YA) system provides dry, oil-free compressed air for the operation of pneumatic control isolation valves, instrument consoles, backfilling of machines, and other air needs at a nominal pressure of 100-110 pounds per square inch gauge (psig). The compressed air contains less than 0.05 parts per million (ppm) oil, with a moisture content equal to

or less than a 0^o Fahrenheit (F) dew point at atmospheric pressure, and contains no particulate greater than 10 microns.

YA for the Lead Cascade is supplied through a header in the service module to the centrifuge machines. YA is also required by the PV, EV and environmental control systems, and is used in the CTTF.

1.1.3 Hazardous Materials Storage

Large quantities of hazardous materials are not present in the Lead Cascade area. Only small quantities of chemicals and materials (e.g., acetone, solvents, oils) are used in the X-7726 and X-3012 buildings, primarily for assembly and maintenance activities. Storage of the chemicals and materials is in approved containers. Those items are listed in the Hazardous Material Inventory Control System. These materials are appropriately reported annually to the Federal and State EPA as required by the *Superfund Amendments Reauthorization Act* (SARA Sections 312 and 313 reports).

The Licensee complies with all requirements for generators of hazardous and mixed waste. The State of Ohio has adopted a federal exemption[1] to the hazardous waste rules that is available under 40 CFR Part 266 Subpart N.

1.1.4 Roadways and Railroads

The PORTS reservation is serviced by two major four lane highways: U.S. Route 23, traversing north-south, and State Route 32/124, traversing east-west. PORTS is situated approximately three and one half miles from the intersection of U.S. Route 23 and State Route 32/124. Primary ingress and egress from the reservation to these major roadways is by the Main Access Road, which connects to U.S. Route 23. The Main Access Road connects to the Perimeter Road, which encircles the PORTS reservation. There are access roads that intersect the Perimeter Road from all four sides. However, vehicular traffic access to the Perimeter Road and PORTS reservation area adjacent to the Lead Cascade are controlled by PORTS Protective Forces. Service roads throughout the reservation connect to Perimeter Road with access to the facility controlled through security portals. The reservation roadways are depicted in Figure 1.1-1.

The Norfolk Southern rail line is connected to the CSX Transportation Inc. main rail system by a rail spur entering the northern portion of the reservation. The CSX Transportation Inc. system also provides access to other rail carriers. Several track configurations are possible within the site. The Lead Cascade is also connected to the existing rail configuration. The railroads are also depicted in Figure 1.1-1.

1.1.5 Site Boundary

PORTS is located approximately one and one half miles east of U.S. Route 23 on approximately 3,708 acres of DOE-owned land. The area around the site is sparsely populated,

[1] At the time of submission of this license application, the State of Ohio was in the process of adopting a federal exemption to the hazardous waste rules that is available under 40 CFR Part 266 Subpart N. This exemption would delegate the Federal EPA's authority over mixed waste to the NRC, subject to certain conditions. The State of Ohio can adopt this exemption by amending the Ohio EPA regulations to adopt the federal exemption by reference.

with the nearest residential center located approximately four miles to the north. The Lead Cascade facilities are located in the western portion of the reservation. Proximity to the nearest member of the public is about 900 meters (m).

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 1.1-1
PORTS Reservation

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 1.1-2
Lead Cascade Facility Layout

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 1.1-3
CTTF Assembly Stand

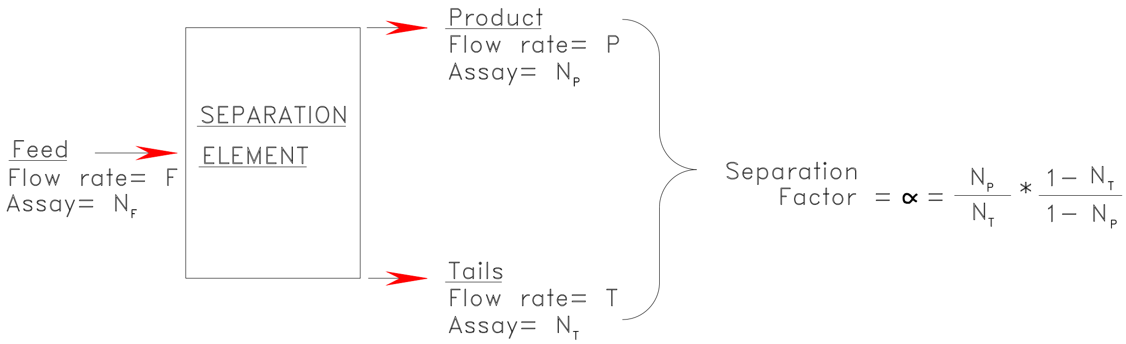

Figure 1.1-4
Separation Element

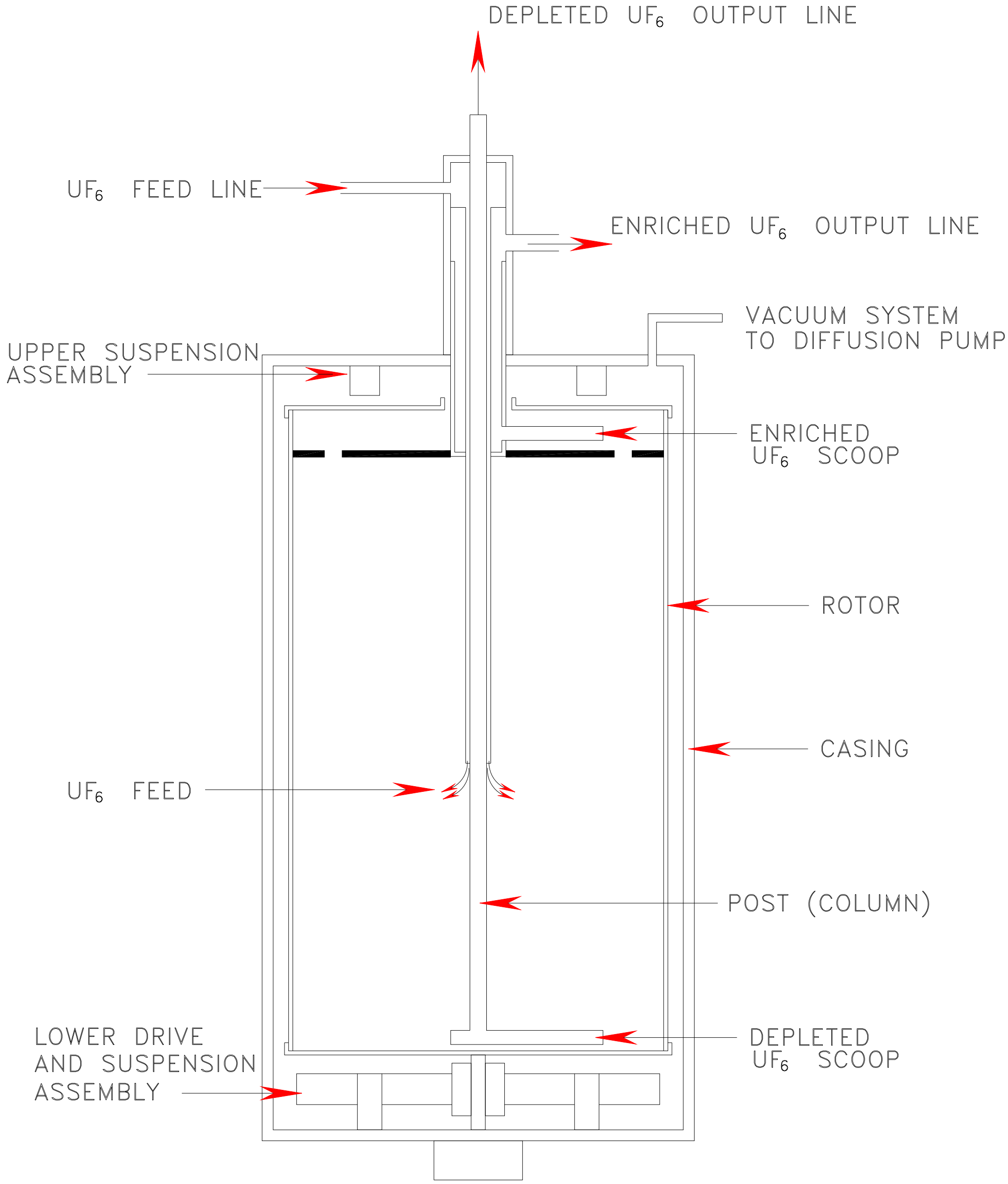

Figure 1.1-5
Centrifuge Schematic

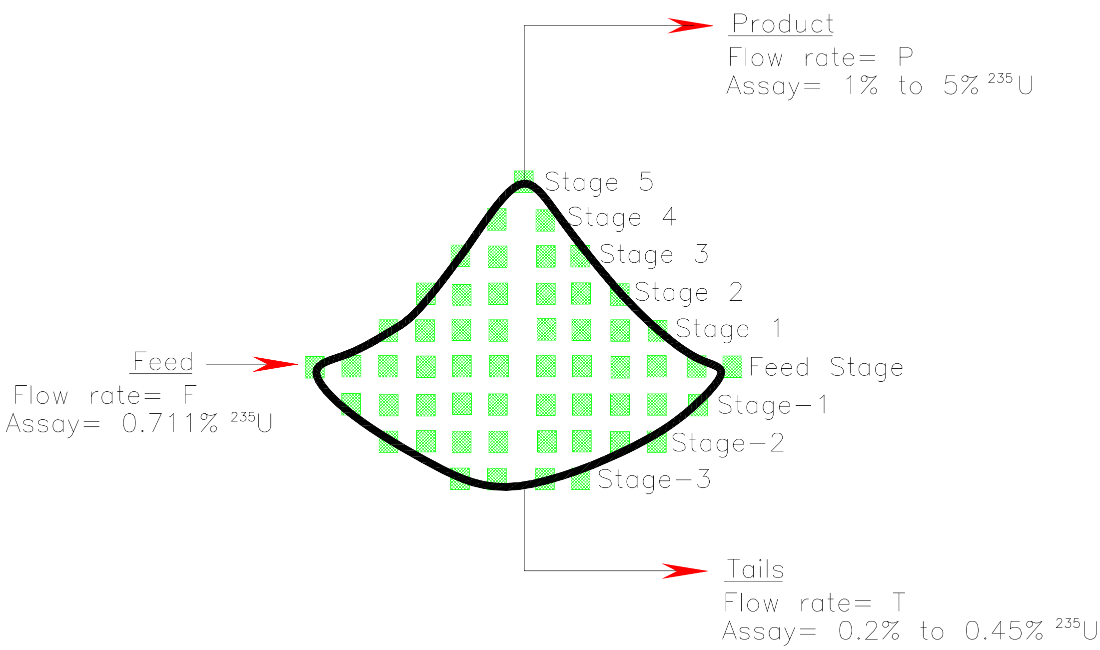

Figure 1.1-6
Example Cascade Schematic

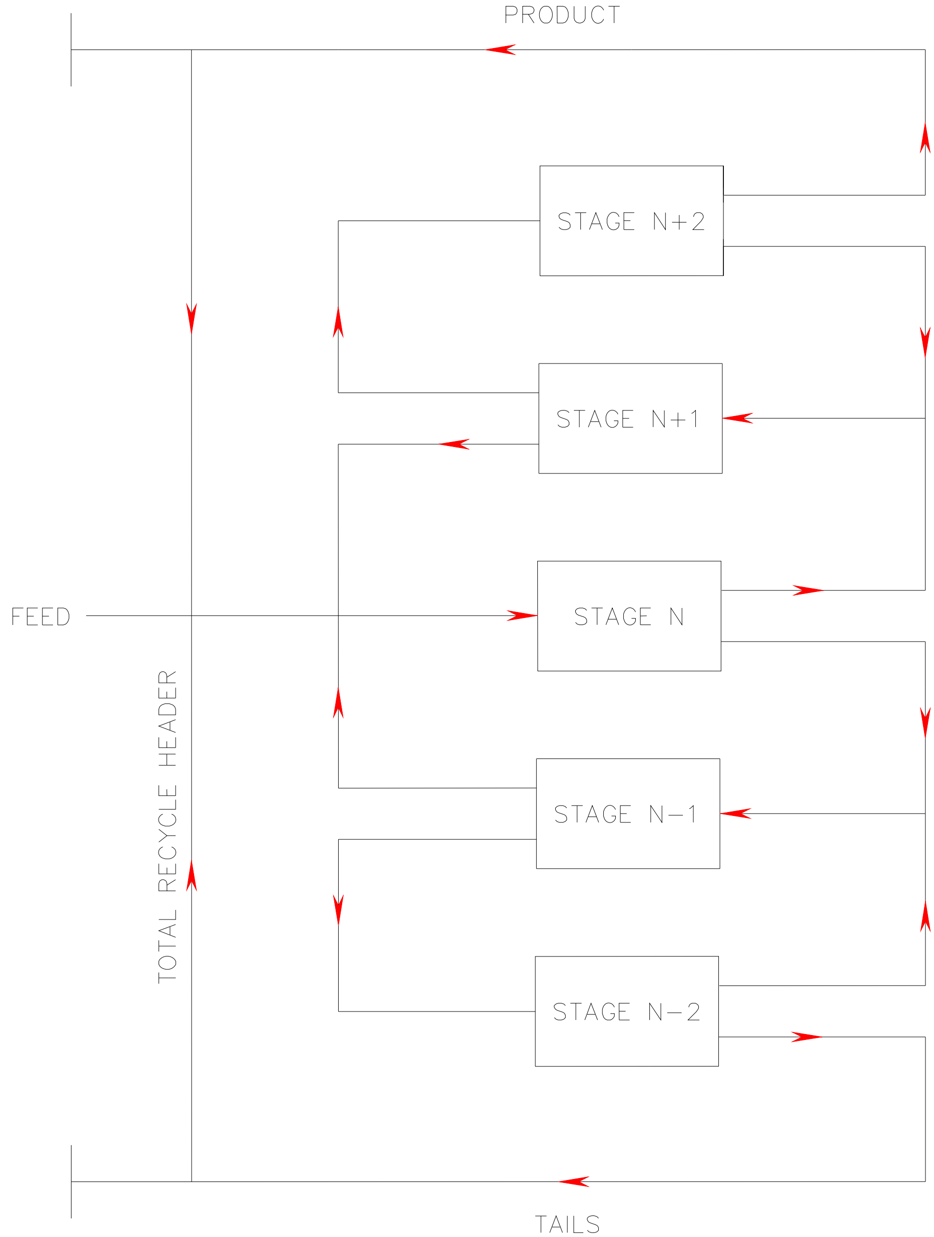

Figure 1.1-7
Example Cascade and Stage Flow Schematic

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 1.1-8
Lead Cascade Systems Interfaces

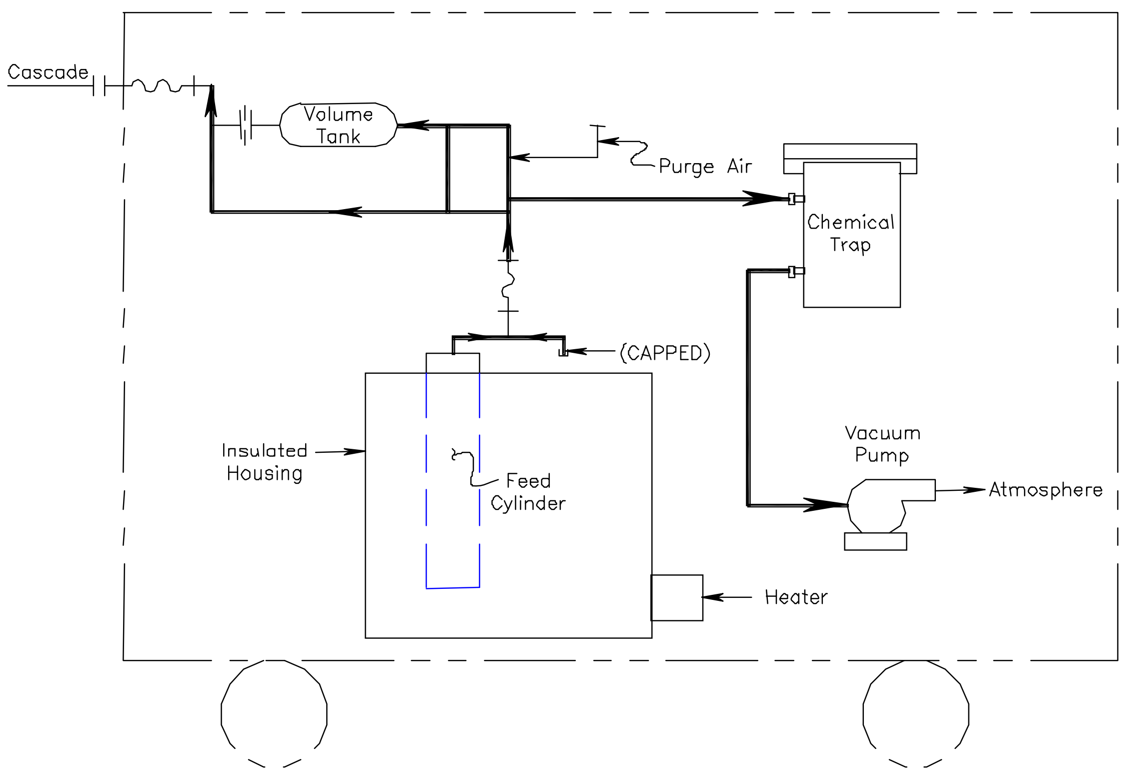

Figure 1.1-9
Feed Cart Flow Diagram

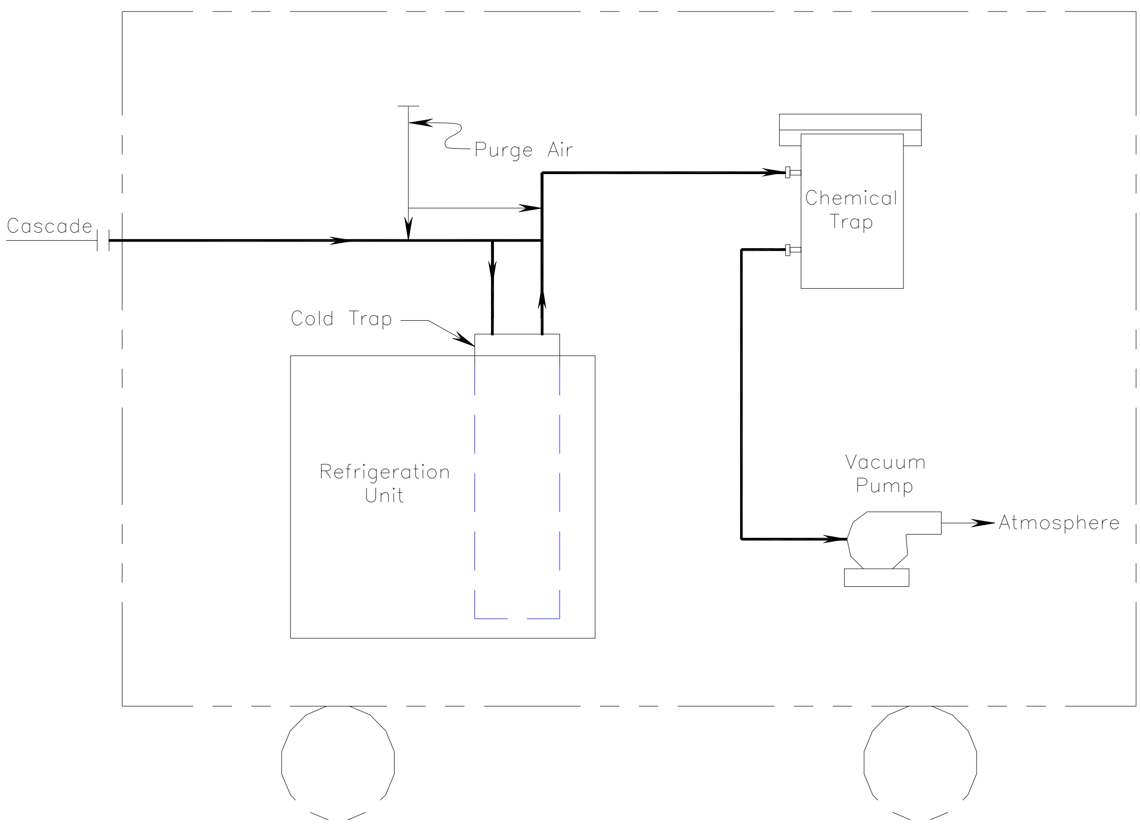

Figure 1.1-10
Dump Cart Flow Diagram

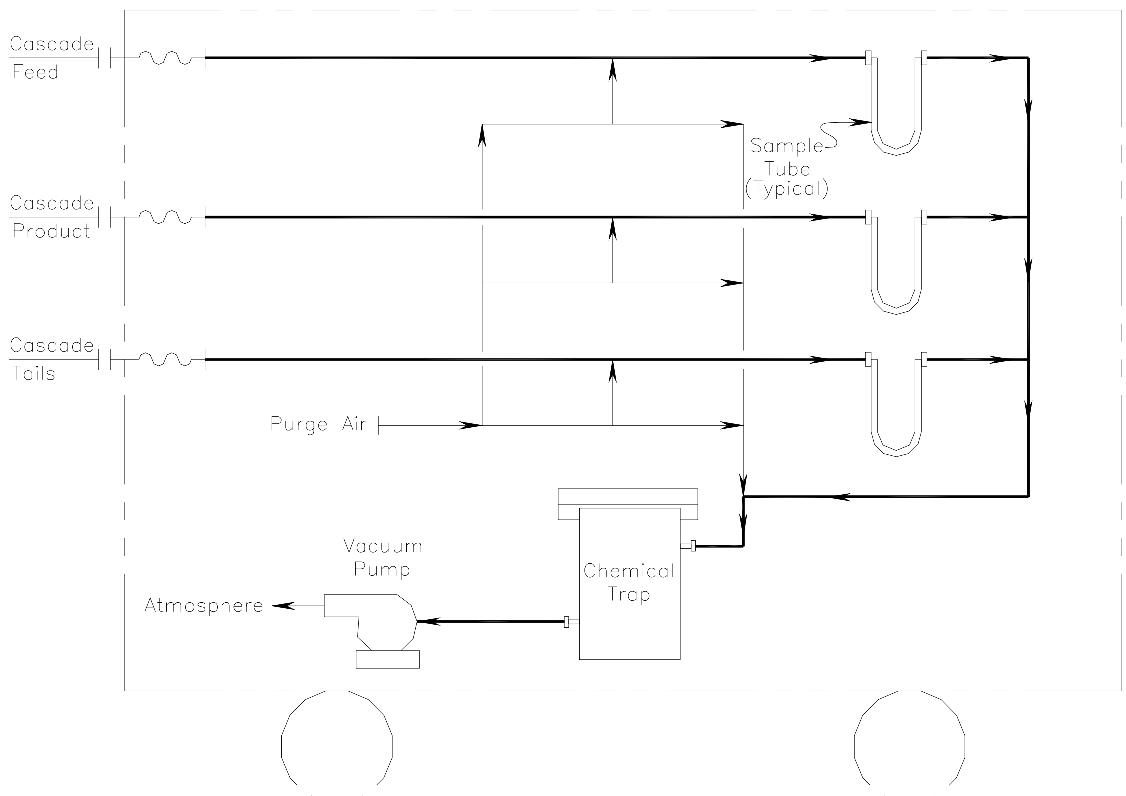

Figure 1.1-11
Sample Cart Flow Diagram

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application.

Figure 1.1-12
Purge Vacuum and Evacuation Vacuum System Schematic

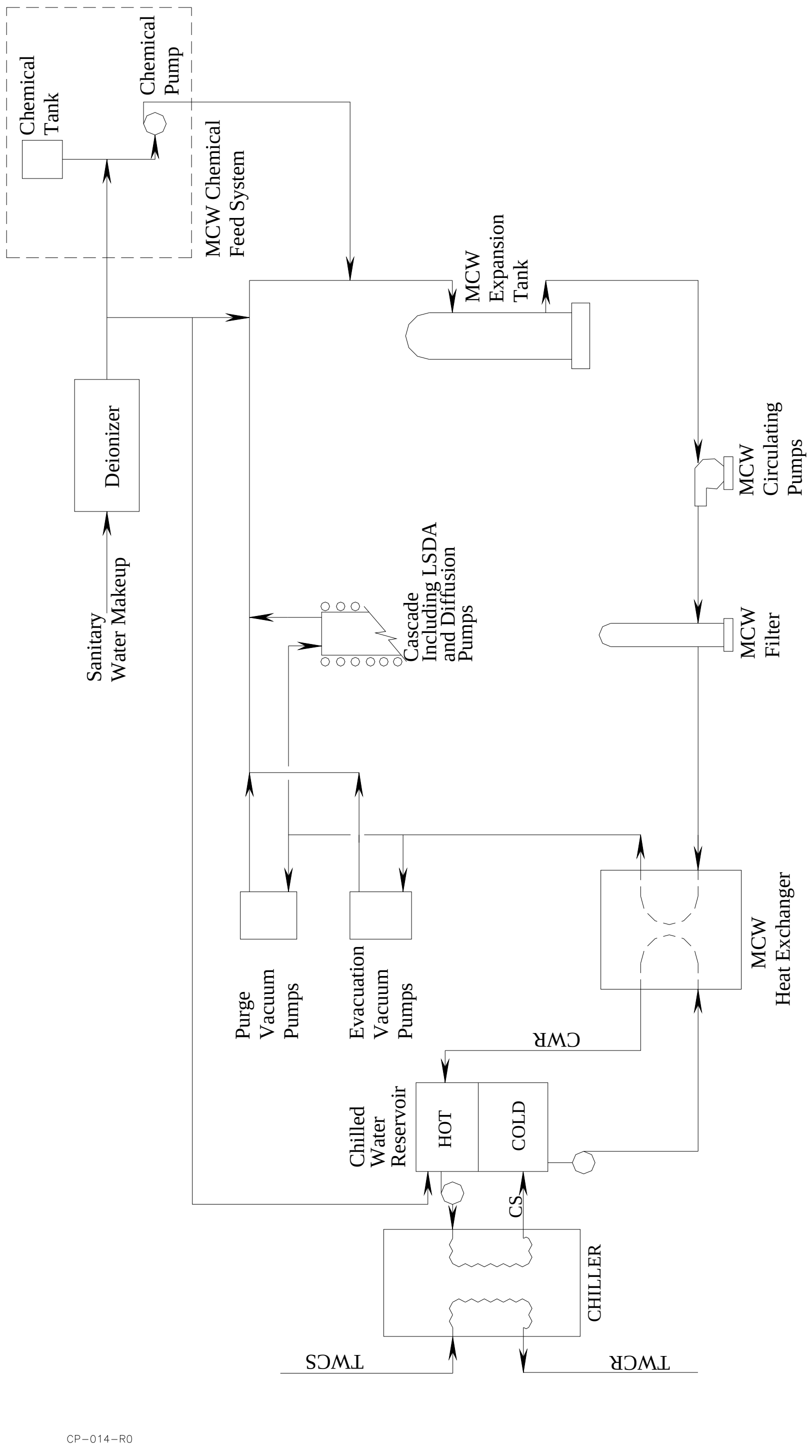

Figure 1.1-13
Machine Cooling Water System Flow Schematic

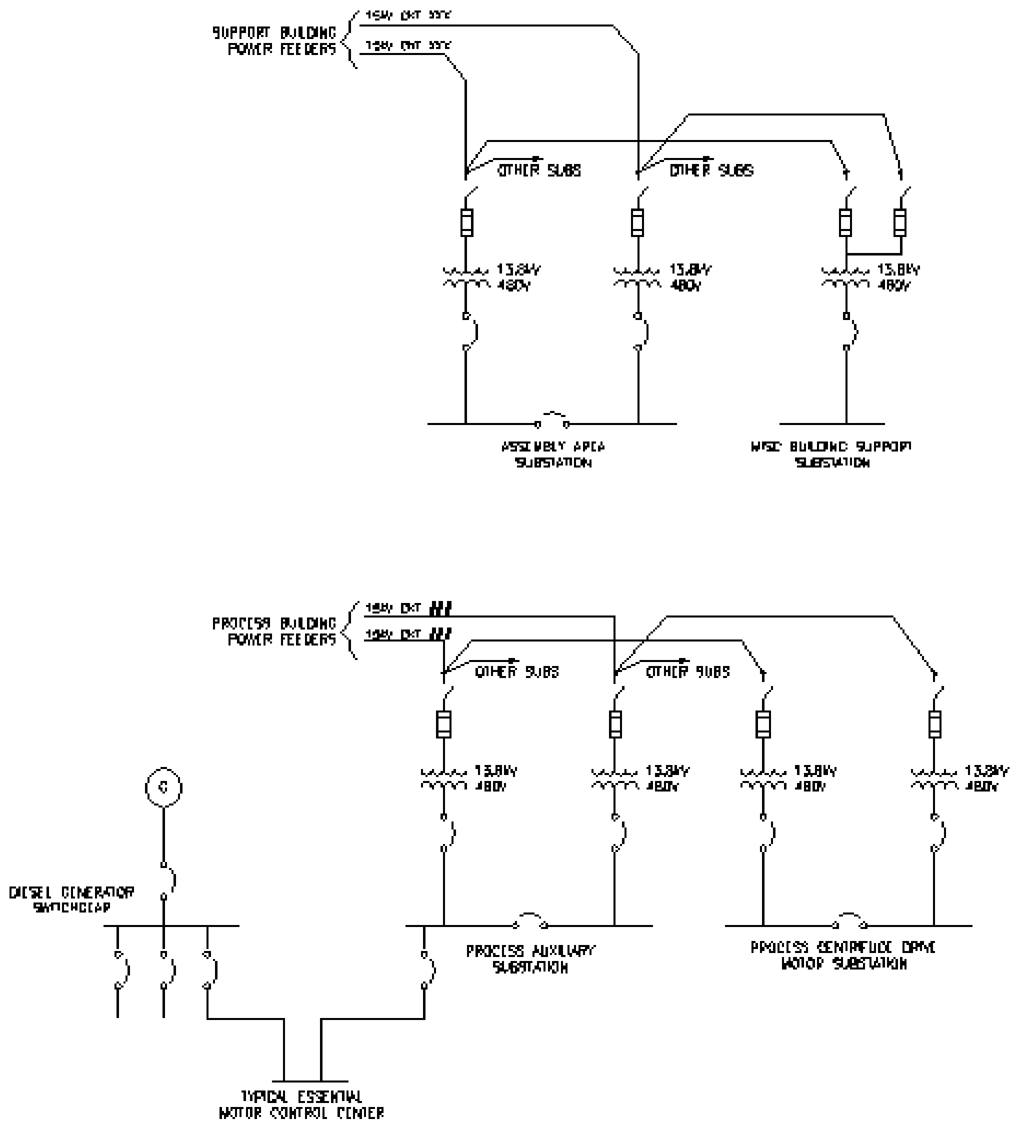

Figure 1.1-14
Lead Cascade Electrical One Line Diagram

1.2 Institutional Information

USEC Inc. is the applicant for the American Centrifuge Lead Cascade Facility license to possess and use special nuclear, source, and by-product material.

1.2.1 Corporate Identity

USEC is a global energy company and the world's leading supplier of enriched uranium fuel for commercial nuclear power plants. USEC, including its wholly owned subsidiaries, was organized under Delaware law in connection with the privatization of the Corporation.

USEC is responsible for the design, quality assurance, refurbishment/construction, manufacturing, testing, startup, operation, and maintenance of the Lead Cascade that is located at PORTS. PORTS is being maintained in Cold Standby status by the Corporation (a wholly owned subsidiary of USEC) under contract to the DOE.

USEC's principle 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 all outstanding shares of USEC. The principal officers of USEC are listed below and all are citizens of the United States.

John K. Welch, President and Chief Executive Officer
W. Lance Wright, Senior Vice President
Philip G. Sewell, Senior Vice President
Robert Van Namen, Senior Vice President
Peter B. Saba, Senior Vice President
Christine M. Ciccone, Senior Vice President
John C. Barpoulis, Senior Vice President and Chief Financial Officer

The NRC has issued Certificates of Compliance to the Corporation to operate the Paducah and Portsmouth Gaseous Diffusion Plants (Docket Numbers 70-7001 and 70-7002, respectively). Consistent with the requirements in 10 CFR 76.22 and in connection with the issuance of these Certificates, the NRC has determined that USEC is neither owned, controlled, nor dominated by an alien, a foreign corporation, or a foreign government.

The mailing address for the Lead Cascade Project is:

USEC Inc.
Lead Cascade Project
P. O. Box 628
Piketon, Ohio 45661

1.2.1.1 Facility Site Location

The Lead Cascade is located at PORTS. PORTS is located at 39º 00' 30" north latitude and 83º 00' 00" west longitude measured at the center of the plant, on an approximate 3,708-acre federally owned reservation in Pike County, Ohio. The largest cities within an approximate 50-mile radius are Portsmouth, Ohio, located approximately 27 miles to the south, and Chillicothe, Ohio, located approximately 27 miles to the north. PORTS occupies a security-fenced Controlled Access Area about one and one half miles east of U.S. Route 23 and two miles south of State Route 32, and two miles east of the Scioto River.

USEC, through its subsidiary (the Corporation), leases PORTS from the DOE. The Lead Cascade is within the space leased by USEC from the DOE. USEC and its agents will conduct USEC activities within the Lead Cascade facilities and access and egress thereto, in accordance with this license application.

1.2.1.2 Other Reservation Activities

PORTS operates in accordance with a NRC Certificate of Compliance issued pursuant to 10 CFR Part 76 requirements. The Corporation's operations include:

- Performing uranium deposit removal activities in the cascade facilities; and

- Activities necessary to support DOE decontamination and demolition of the gaseous diffusion plant facilities.

Source and byproduct material sources used at the Lead Cascade are authorized utilizing the PORTS NRC Certificate of Compliance.

The Corporation is constructing the American Centrifuge Plant (ACP) to utilize centrifuge technology developed under the Lead Cascade. ACP construction activities will occur in various facilities that house and surround the Lead Cascade.

In addition to the Corporation's operations, DOE is engaged in restoration, decontamination, and demolition activities in a number of locations on the reservation and utilizes contractors and sub-contractors to perform this work. DOE self-regulates DOE activities conducted in non-leased and leased areas in accordance with applicable DOE requirements. Other DOE activities for consideration include, DOE has constructed and plans to operate a depleted uranium hexafluoride (DUF_6) Conversion Facility on the site north of the X-7725 Recycle/Assembly Building. Additionally, the Ohio National Guard maintains an area on the PORTS reservation for the maintenance, reconditioning, and storage of equipment. No ordnance is permitted. The activities are accomplished in and around the X-751 facility, located on the south end of the site.

1.2.2 Financial Qualifications

The Lead Cascade is financed by USEC. The refurbishment would employ an average of 25 workers for a 13-month period. Upon start-up, approximately 45 full-time employees staff the Lead Cascade for the limited period of planned operations. The Lead Cascade refurbishment and operating costs are documented in Appendix D of the Environmental Report for the American Centrifuge Lead Cascade Facility.

As reported in USEC's fiscal year 2002 Annual Report (Reference 1), based on customers' estimates for enrichment requirements and certain other assumptions, including estimates of inflation rates, at June 30, 2002, USEC had long-term requirements contracts aggregating $4.5 billion through fiscal 2011 (including $2.7 billion through fiscal 2005). Net income amounted to $16.2 million (or $.20 per share) in fiscal 2002.

USEC is the sole provider of funds for the construction and operation of the Lead Cascade. Expenditures related to the Lead Cascade, including any related cost over-runs, will be paid out of existing and projected cash flows from on-going operations. In light of its strong cash flow and cash position, and considering corporate obligations over the life of the project, USEC has confidence in its ability to complete construction and operation of the Lead Cascade with the financial assets and mechanisms available to it.

1.2.3 Type, Quantity, and Form of Licensed Material

The type, quantity, and form of NRC-regulated source and special nuclear material are shown in Table 1.2-1.

1.2.4 Authorized Uses

The Lead Cascade enriches UF_6 up to 10 wt. percent ^{235}U. The Lead Cascade is operated on recycle where the enriched product stream is recombined with the depleted stream prior to being re-fed to the cascade. No product withdrawals are made from the Lead Cascade except for samples taken for laboratory analysis. Withdrawal of small quantities of depleted material may also be performed on an infrequent basis for operational considerations, with subsequent addition of feed to the cascade. The Lead Cascade may possess up to 250 kg UF_6. The specific authorized uses for each class of NRC-regulated material are shown in Table 1.2-2.

1.2.5 Special Exemptions or Special Authorizations

Similar to GDP operations, the following exemption to the applicable 10 CFR Part 20 posting and labeling requirements are identified in Section 4.8.1 of this license application:

- Five, eight, and twelve-inch uranium cylinders are routinely transported between facility locations and/or storage areas at the Lead Cascade and are readily identifiable due to their size and unique construction. These are not routinely labeled as radioactive material. The transportation of UF_6 cylinders and UF_6 sample containers are constantly attended by qualified Radiological Workers during movement.

The following exemption to the applicable 10 CFR 70.50 reporting requirement is identified in Section 11.6.6 of this license application:

- The 10 CFR 70.50(c)(2) reporting criteria requires that the Licensee submit a written 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 Licensee requests NRC approval to submit the required written reports within 60 days of the initial notifications.

The following Special Authorization has been identified in this license application:

- 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.

1.2.6 Security of Classified Information

The Licensee is required by 10 CFR 70.22(m) to submit, as part of its application for a license for the Lead Cascade, a plan describing the facility's proposed security procedures and controls, as set forth in 10 CFR Part 95, for the protection of classified matter. The Licensee satisfied this requirement by submission of its plan for the protection of classified matter as Chapter 2 of the Security Program.

Table 1.2-1
Lead Cascade Possession Limits

Type of Material	Atomic Number	Physical State	Chemical Form	Possession Limit	Description
A. Source Material	92	Solid and gas	UF_6, UF_4, UO_2F_2, oxides, and other compounds	250 kg UF_6 169 kg U	Uranium (including natural and depleted) and daughter products and process contaminants and wastes
B. Special Nuclear Material[a]	92	Solid and gas	UF_6, UF_4, UO_2F_2 , oxides, metal, and other compounds	700 g ^{235}U	Uranium enriched in isotope 235 up to 10 percent by weight, uranium daughter products and process contaminants and wastes, to include: (1) instrument calibration and check sources, or (2) material that may be in process and/or held up in facilities and equipment from Lead Cascade operations
	94	Sealed source		0.5 Ci	Instrument calibration sources, NDA

Table 1.2-1 (continued)
Lead Cascade Possession Limits

Type of Material	Atomic Number	Physical State	Chemical Form	Possession Limit	Description
Information Deleted					

a. See 10 CFR Part 70 definitions. Special nuclear material means: (1) Plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to the provisions of Section 5l of the Act, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material.

Table 1.2-2
Authorized Uses of NRC-Regulated Materials

Material Class	Authorized Use
A. Source Material, Element 92	1. Enrichment of uranium up to 10 percent enrichment by weight ^{235}U
	2. Receipt, storage, inspection, and acceptance sampling of cylinders containing uranium
	3. Filling and storage of cylinders of natural uranium and uranium depleted in ^{235}U
	4. Cleaning and inspection of cylinders used for the storage and transport of source material
	5. Storage of process wastes containing uranium and other contaminants and decay products
	6. Process, characterize, package, or store low-level radioactive and mixed wastes
	7. Radiation protection, process control and environmental sample collection, analysis, and operation checks
	8. Maintenance, repair, and replacement of process equipment
	9. Heating cylinders and feeding contents into the centrifuge process
	10. Transfer between cylinders

Table 1.2-2 (continued)
Authorized Uses of NRC-Regulated Materials

Material Class	Authorized Use
B. Special Nuclear Material	1. Filling and storage of cylinders containing uranium enriched up to 10 percent by weight ^{235}U
	2. Nondestructive testing and analyses of process streams
	3. Cleaning and inspection of cylinders used for the storage and transport of special nuclear material
	4. Storage of process wastes containing uranium and other contaminants and decay products
	5. Process, characterize, package, or store low-level radioactive and mixed wastes
	6. Radiation protection, process control and environmental sample collection, analysis, instrument calibration, and operation checks
	7. Maintenance, repair, and replacement of process equipment
	8. Heating cylinders and feeding contents into the centrifuge process

1.3 Site Description

This section describes the Lead Cascade's location and description, nearby roadways and bodies of water, and significant geographical features.

The Lead Cascade is located on DOE-owned land in rural Pike County, a sparsely populated area in south central Ohio. Specifically, the Lead Cascade is located on the PORTS reservation in the former GCEP facilities. (Figure 1.1-1) The facilities are leased (through the United States Enrichment Corporation) from the DOE. The PORTS reservation has been studied and characterized extensively by both DOE and USEC.

1.3.1 Geography

The PORTS reservation is on the east side of the Scioto River approximately equidistant between Portsmouth and Chillicothe, Ohio.

The Scioto River Valley is one mile west of the site. The Scioto River, approximately two miles west of the site, is a tributary of the Ohio River, and their confluence is approximately 25 miles south of the PORTS reservation. With the exception of the Scioto River floodplain, which is farmed extensively, the area around the site consists of marginal farmland and forested hills. The only other body of water located near the site is Lake White, located approximately six miles north of the site.

Two major four lane highways: U.S. Route 23, traversing north-south, and State Route 32/124, traversing east-west, service the PORTS 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 aircraft, is located approximately 15 miles southeast near Minford, Ohio, and the Pike County Airport, located north of Waverly, is a small facility for private planes.

The entire PORTS reservation on which the Lead Cascade facilities are located 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. The reservation boundary is the controlled area boundary specified in 10 CFR 70.61(f). Most buildings and activities at the site (including the Lead Cascade facilities) are located within the next level of control, a Property Protection Area or Controlled Access Area (CAA), both surrounded by a security fence. Access to this fenced area is gained only with approved identification. In addition, the Lead Cascade is located within its own CAA. See the Security Program document for the location of the CAA/security fence. A topographic map of the PORTS reservation is provided in Figure 1.3-1.

1.3.2 Demographics

The PORTS reservation is located in Pike County, which is primarily rural in nature. With the exception of the Scioto River floodplain, which is farmed extensively, the area around the site consists of marginal farmland and forested hills. The remaining counties in the vicinity are also largely rural in character, except near the towns of Portsmouth in Scioto County and Chillicothe in Ross County.

1.3.2.1 Area Residential Population

The nearest residential center and the closest town to the site is Piketon, located in Pike County about four 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 eight miles north of the site, with a population of 4,433 in 2000. Chillicothe, in Ross County about 27 miles north, is the largest population center in the Region of Influence with a population of 21,796 in 2000. Other population centers include Portsmouth, about 27 miles south in Scioto County, and Jackson, about 26 miles east in Jackson County, with populations of 20,909 and 6,184 in 2000, respectively. Table 1.3-1 presents historic and projected population in the Region of Influence and the state (Reference 2). The total population within the five-mile radius of the site is 5,836 (Figure 1.3-2).

1.3.2.2 Significant Transient and Special Populations

In addition to the residential population, there are institutional, transient, and seasonal populations in the PORTS area.

1.3.2.2.1 Schools

There are a number of educational institutions inside a five-mile radius of the PORTS site. All of the Scioto Valley Local School District's (SVLSD) schools are within the five-mile radius. They are the Piketon High School and Junior High School, located in the same building with 635 students and 66 staff, Zahn's Corner Middle School with 366 students and 44 staff, and Jasper Elementary School with 517 students and 49 staff. In addition to the SVLSD there is the Pike County Career Technology Center with 439 vocational high school students, 100 adult education students, and 79 staff. There are also two public preschools with daycare; the Early Childhood Family Center with 35 students and 32 staff, and the Pike County Community Action Committee with 96 students and 148 staff, and a private pre and elementary school, Miracle City Academy, with 13 students and 4 staff. The locations and student-occupancies of these facilities are shown in Figure 1.3-3 (Reference 3).

1.3.2.2.2 Hospitals and Nursing Homes

Pike Community Hospital is the hospital closest to the site, located approximately 7.5 miles north of the facility off of State Route 104 south of Waverly. The hospital facility has 25 licensed beds, 270 total staff, and operates at full capacity. Adena Health System operates an urgent care facility located in Waverly approximately 1 mile north of the hospital. The Waverly Family Health Center is located next to the Pike Community Hospital, and the Piketon Family Health and Dental Center is located in Piketon.

There are two licensed nursing homes in the Piketon area, the Piketon Nursing Center with 46 patients and 46 staff and the Pleasant Hill Manor with 193 patients and 220 staff, and a home for the mentally retarded in Wakefield, Friends of Good Shepherd Manor, with 51 residents and 100 staff. Figure 1.3-3 depicts these medical and nursing facilities and shows the number of beds per facility (Reference 3).

1.3.2.2.3 Recreation Areas and Recreational Events

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

Offsite 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 (Reference 4).

Usage of Lake White State Park (Figure 1.3-3), located approximately six miles north of the site, is occasionally heavy and concentrated on the 107 acres of land closest to the lake. Most of the land surrounding the lake is privately owned. The 350-acre Lake White offers recreations, such as, boating, fishing, and swimming. There are 23 campsites for primitive overnight camping (Reference 5).

1.3.2.3 Uses of Nearby Land and Waters

Land within five miles of the site is used primarily for farms, forests, and urban or suburban residences. About 25,430 acres of farmland, including cropland, wooded lot, and pasture, lie within five miles of the site. 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 PORTS 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 Lead Cascade operations.

Approximately 24,400 acres of forest lie within five miles of the site. This includes some commercial woodlands and a very small portion of Brush Creek State Forest.

No known public or private water supply draws from the Scioto River (Reference 4).

1.3.3 Meteorology

This section provides a meteorological description of the site and its surrounding area. The purpose is to provide meteorological information necessary to understand the regional weather phenomena of concern for the site operations and to understand the basis for the dispersion analyses performed (Reference 4).

1.3.3.1 Regional Climatology

Located west of the Appalachian Mountains, the region around the site has a climate essentially continental in nature, characterized by moderate extremes of heat and cold and wetness and dryness (Reference 4). July is the hottest month, with an average monthly temperature of 74°F, and January is the coldest month with an average temperature of 30°F. The highest and lowest daily temperatures from 1951 to 1980 were 103 and –25°F on July 14, 1954, and February 3, 1951, respectively.

Moisture in the area is predominantly supplied by air moving northward from the Gulf of Mexico (Reference 4). Precipitation is abundant from March through August and sparse in October and February. The average annual precipitation at Waverly, Ohio, for the period from 1951 to 1980 was 40.4 inches (in.) (Reference 4). The greatest daily rainfall during this period was 3.38 in., occurring on June 26, 1971. Snowfall occurrence varies from year to year, but is common from November through March. The average annual snowfall for the area is about 22 in., based on the 1951-1980 data. During that time period, the maximum monthly snowfall was 25.4 in., occurring in January 1978.

Occasionally, heavy amounts of rain associated with thunderstorms or low pressure systems will fall in a short period of time. The U.S. Weather Bureau has published values of the total precipitation for durations from 30 minutes to 24 hours and return periods from 1 to 100 years. The results for the geographic locale including the site are summarized in Table 1.3-2. A local drainage analysis for extreme storms at the site has been performed (see Reference 4).

The predominant winds at the site blow from the south or southwest and at times from the north. The average wind speed is about 5 miles per hour (mph). On the average, from 1953 to 1989, 14 tornadoes per year were reported in Ohio, but the total varies widely from year to year (e.g., 43 in 1973 and 0 in 1988). Pike County, where the PORTS reservation is located, had two tornadoes during the 20-year period from 1953 to 1972 (Reference 4).

1.3.3.2 On-Site Meteorological Measurements Program

Since January 1995, a new 200-ft (60-m) tower has been in use. It is equipped with instrument packages at the 33-, 98-, and 200-ft (10-, 30-, and 60-m) levels. In addition, ground-level instrumentation measures solar radiation, barometric pressure, precipitation, and soil temperatures at 1- and 2-ft depths.

1.3.3.3 Local Meteorology

Hourly temperatures at the 33- and 105-ft (10- and 32-m) levels above the ground were recorded at the site meteorological tower before 1995. At each level, 8,555 of the possible 8,760 data points are available. The seasonal temperature variation and the daily temperature fluctuations are consistent with long-term averages. The two sets of temperature readings at the site meteorological tower are highly correlated, as one would expect. Since January 1995, temperatures at the 33-, 98-, and 200-ft (10-, 30-, and 60-m) levels have been measured at the new tower.

Of the 8,760 possible hourly wind speed and wind direction data for 1993, 8,430 are available for wind speed and 8,423 for wind direction. The average wind speeds were 3.7 and 6.0 mph at 33- and 105-ft (10- and 32-m) levels, respectively. Wind roses at 33- and 105-ft (10- and 32-m) at the site constructed from the 1993 data are compared in Figure 1.3-4.

Wind damage at the plant is not likely to result in a significant release of hazardous material. Wind is more likely to cause exterior damage to the buildings without extensive damage internally. In addition, high winds will rapidly disperse any hazardous material released as well as reduce exposure times down wind. Therefore, the risk of serious injuries and/or deaths is substantially low for high winds. The Lead Cascade facilities are designed for a wind loading criteria meeting basic wind speeds of 90 mph at 30 ft above ground.

Tornadoes do occur in Southern Ohio; however, specific analyses of the frequency of tornadoes in the region show that they are rare. The actual damage expected to cascade internal equipment and structures is expected to be minimal on the cascade floor due to the large reservoir of air between the building roof and the cascade floor. The process building containing the Lead Cascade is designed with a tornadic wind of 100 mph and a rate of atmospheric change of 7 pounds per square foot (psf)/second for a duration of 3 seconds. Other Lead Cascade facilities are designed for a wind velocity of 90 mph at 30 feet above ground except standard prefabricated buildings used as offices, warehouses and nonessential facilities less than 30 feet in height. This last category of facilities is designed for a horizontal wind pressure of 20 psf.

Additionally, the Lead Cascade facilities are designed with a ground snow load of 20 psf.

Because PORTS is not a coastal location, the effects of hurricanes are not considered other than increased rainfalls as remnants of the storm affected weather patterns in the upper Ohio River Valley.

Severe storms can and are likely to produce lightning strikes, which can interrupt and cause a partial power failure. However, the buildings are heavily grounded and some have installed lightning protection. PORTS is in an area that has less than 40 days annually with a thunderstorm. PORTS is at a “moderate” risk value of loss due to lightning strikes. Lightning has not been a problem for these structures, since initial construction in the mid-1980s. The power systems are designed to handle lightning strikes.

1.3.4 Surface Hydrology

This section describes the surface hydrology on and around the PORTS site.

1.3.4.1 Hydrologic Description

The significant surface streams and waterways affecting the PORTS reservation are discussed in this section.

1.3.4.1.1 Scioto River Basin

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 (RM) 356.5. The slope of the Scioto River channel averages about 1.7 ft/mile between Columbus and Portsmouth (Reference 4).

Upstream retarding basins are located on tributaries throughout the Scioto River basin. The upstream retarding basin nearest the site forms Lake White along Pee Pee Creek, about six miles north of the site (Figure1.3-5). The spillway of the reservoir is located at an elevation of 567 ft, while the roadway along the top of the dam is at an elevation of 577 ft (Reference 4). Pee Pee Creek empties into the Scioto River south of Waverly at RM 40.

The U. S. Geological Survey (USGS) has collected stream-flow data for the Scioto River at Higby, Ohio, since 1930. The gauging station is located approximately 13 miles north of the site at RM 55.5. The drainage area of the Scioto River basin above Higby is 5,130 square miles. The river flows measured at Higby from 1930 to 1991 range 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. The consecutive seven-day minimum discharge of record is 255 cfs, which occurred during October 19-25, 1930 (Reference 4).

Water in the vicinity of the site is available from Lake White, the Scioto River, and groundwater supplies (Reference 4). Most of the water used is taken from groundwater. Three municipal water supply facilities are located in the segment of the Scioto River between Higby and the confluence with the Ohio River (and all three use groundwater wells). Both Waverly and Piketon, located at RM 40 and 34, respectively, use groundwater wells. The city of Portsmouth uses water from the Ohio River through an intake at the Ohio River at RM 350.8, which is 5.7 miles upstream from the mouth of the Scioto River (Reference 4).

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 (X-605G, X-608A, X-608B, and X-6609) have the capacity to supply reliably between 36.4 and 40.2 cfs.

1.3.4.1.2 PORTS Reservation Area

The site is located about 2.5 miles east of the confluence of the Scioto River and Big Beaver Creek near RM 27.5 (Figure 1.3-5). The plant site occupies an upland area bounded on the east and west by ridges of low-lying hills that have been deeply dissected by present and past drainage features. The plant nominal elevation is 670 ft, which is about 130 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.

Both Big Beaver and Little Beaver Creeks receive runoff from the northeastern and northern portions of the site. Little Beaver Creek, the largest stream on the property, flows northwesterly just north of the main plant area (Figure 1.3-5). It drains the northern and northeastern parts of the main plant site before discharging into Big Beaver. About two miles from the confluence of the two creeks, Big Beaver Creek empties into the Scioto River at RM 27.5 (Figure 1.3-5). Upstream from the plant, Little Beaver Creek has intermittent flow throughout the year.

In the southeast portion of the site, the southerly flowing Big Run Creek (Figure 1.3-5) is situated in a relatively broad, gently sloping valley where significant deposits of recent alluvium have been laid down by the stream (Reference 4). This intermittent stream receives overflow from the south holding pond (X-230K), which collects discharge of storm sewers on the south end of the plant site. Big Run Creek empties into the Scioto River about five miles downstream from the mouth of Big Beaver Creek (Figure 1.3-5).

Two unnamed intermittent streams drain the western portion of the plant site (Figure 1.3-5). The stream in the site's southwest portion flows southerly and westerly in a narrow, steep-walled valley with little recent alluvium. It drains the southwest corner of the facility via the southwest holding pond. The stream near the west central portion of the plant site flows northwesterly and receives runoff from the central and western part of the site via the west drainage ditch. Both unnamed streams flow directly to the Scioto River and carry only storm water runoff (Reference 4).

Little Beaver Creek receives 39 percent of the total PORTS reservation effluents, Big Run Creek, 9 percent, and the two unnamed tributaries, 25 percent. The remaining 27 percent is discharged directly to the Scioto River through two pipelines. Treated effluents from a sanitary sewage plant are conveyed about two miles to the Scioto River via a 15-in. vitreous clay sewer line at Outfall 003; blowdown from the recirculating cooling water system enters the Scioto via Outfall 004 (Reference 4).

1.3.4.1.3 Site and Facilities

The PORTS reservation nominal elevation is 670 ft, which is about 130 ft above the normal stage of the Scioto River. The top-of-slab floor elevations for the Lead Cascade facilities are at approximately 671 ft. 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 (located in Appendix B of this license application), serves as a hydrologic boundary that prevents storm water runoff from backing up into the Lead Cascade facilities. Once storm water has been discharged onto the outer side of the Perimeter Road to the north, west, and south, the water flows downhill to local creeks and runs. To the east and southeast, the Perimeter Road acts as a diversion dam that directs storm water runoff to Big Run Creek. The northeastern corner of the Perimeter Road protects the main gaseous diffusion plant process buildings from flooding that could occur if the X-611B sludge lagoon dam failed. The relationship of storm water holding ponds, located along the outside of Perimeter Road shown in Figure 1.3-6, to the topographic elevations, indicated in Figure 1.3-7 (located in Appendix A of this license application), emphasizes the overall function of the site surface water drainage system that has been described here (Reference 4).

Water used at the site is supplied by wells sunk into the Scioto River alluvium. The raw water is pumped through a 48-in. waterline from the X-608 Raw Water Pump House and through a 30-in. waterline from X-6609 to the Water Treatment Plant, X-611, located near the northeastern corner of the site just outside the Perimeter Road. Pump House X-608, near the well fields, can also pump water from the Scioto River and is a backup system that is used only when the well systems are unable to produce sufficient water to meet the plant demand. The well fields and the X-608 pump house may expect flooding (Reference 4). Although equipment in X-608 and X-6609 well fields is designed to operate without the effects of flooding, the equipment in the X-608 pumphouse is located above the 571-ft level.

1.3.4.2 Flood History

The average annual discharge at the Higby station for the period of record (1930-1991) is 4,654 cfs, while the maximum discharge of record is 177,000 cfs observed on January 23, 1937. The stage of the 1937 flood was 593.7 ft above mean sea level (amsl). The historical flood stage of the Scioto River next to the site was estimated to be 556.7 ft by using the estimate that the Scioto River drops approximately 37 ft between the Higby gauging station (RM 55.5) and the mouth of Big Beaver Creek (RM 27.5). Elevations for floods (with three recurrence intervals) at the confluence of the Scioto River and Big Beaver Creek (RM 27.5), estimated by the U. S. Army Corps of Engineers, are compared with the site nominal grade elevation in Table 1.3-3 (Reference 4).

Since the site has a nominal elevation of about 670 ft amsl (Figure 1.3-7) and about 113 ft above the historical flood level for the Scioto River in the area, the site has not been affected by flooding of the Scioto River.

1.3.4.3 Probable Maximum Flood

The plant elevation is greater than the maximum historic levels recorded for the Scioto River in the area and the 500-year flood predicted by the U.S. Army Corps of Engineers. However, a calculation of the Probable Maximum Flood (PMF) was also performed. The details

of a method of calculating the PMF are discussed in NRC Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants*. It is based on the drainage area and the location of the watershed involved. The drainage area of the Scioto River basin above Higby is 5,131 square miles and that of the whole basin is 6,517 square miles (Reference 4). The drainage area of the Scioto River above the site (RM 27.5) is between those two values. A conservative estimate for the PMF discharge of the Scioto River at either Higby or the site is approximately 1,000,000 cfs. This value is used as the PMF discharge of the Scioto River at the site, which including the wind/wave activity contribution, would correspond to a flood level of 571 ft, well below the nominal 670 ft elevation of the site.

Two widely accepted probabilistic methods, the log Pearson III distribution and the Gumbel method, have been considered. The 10,000-year flood discharges of the Scioto River at Higby determined with these two methods are 526,000 and 280,000 cfs, respectively. Both of these discharge rates are smaller than that of the PMF. The PMF is, therefore, the bounding event in determining the evaluation basis loads from flooding for the site.

Conservative estimates indicate that the failure of upstream dams would not threaten the safety of the PORTS reservation because of the high nominal plant grade elevation (Reference 4). In addition, the limited storage capacities of the reservoirs, the large stream distances of these dams from the site, and friction and form losses would make the actual wave heights even smaller than the estimated values. Discharges were considered of dam failures at full pool combined with that of either a 25-year flood or one-half of the PMF of the Scioto River. The result involving one-half of the PMF would result in a higher value, which is also somewhat greater than that of the PMF. However, this combined extreme flood would not threaten the safe operations of the site because of the high nominal plant grade elevation, similar to the case of the PMF.

1.3.4.3.1 Effects of Local Intense Precipitation

Storm Intensities and 10,000-Year Storms

The U.S. Weather Bureau has published values of the total precipitation reaching the ground for durations from 30 minutes to 24 hours and return periods from 1 to 100 years (Reference 4). The results for the geographic locale including the site are summarized in Table 1.3-2. Values for 10,000-year storms are extrapolated from smaller duration values using a least-squares method. The rainfall intensity for a given storm listed in Table 1.3-2 can be obtained by dividing the total precipitation by the duration.

To determine whether the influx of rainwater from a 10,000-year storm can be conveyed away from plant structures, the intensity versus duration relation for 10,000-year storms at the site needs to first be established. This was done by adopting an established empirical intensity versus duration relation and using values listed in the last row of Table 1.3-2 and a nonlinear least-squares methodology (Reference 4). The resultant graph is shown in Figure 1.3-8. At small durations, although the intensities are high, the total precipitations are small. At large durations, the reverse is true.

Results for Creeks

The stage-discharge relationships for the five streams draining the site facilities were evaluated using the estimated cross sections and Manning's formula with n = 0.15, a value typical for flood plains and very poor natural channels. The peak runoffs of these streams can be calculated using the natural runoff model and the intensity vs. duration relation shown in Figure 1.3-8. Local flooding for different streams are caused by 10,000-year storms with differing duration values because each watershed drains a basin of a different size (Reference 4). The relatively large differences between nominal plant grade elevation and the calculated flood stage elevations for the five streams clearly indicate that the Lead Cascade facilities would not be inundated by these streams during a 10,000-year storm.

Results for Storm Sewers

In addition to the Manning's formula and the natural runoff model, the urban runoff model and an inflow-outflow balance method (Reference 4) were also used to assess the storm sewers. In each case, the duration that gives maximum peak discharge is determined and used as the 10,000-year storm.

The results indicate that the site would experience local ponding during a 10,000-year storm because the storm sewer system has insufficient capacity to convey the rainwater to the outfalls. The average depth of water around the base of the buildings would range from 3.91 to 5.08 in. The existing storm sewer system would require from approximately 1.8 to 9.9 hours to drain the excess storm water to the outfalls (Reference 4).

The effect of a clogged storm sewer system on the ponding depth has been considered (Reference 4). Because the storm sewer flow is approximately one-fourth of the total 10,000-year storm flow, the overland drainage system is the dominant factor in determining the water depth at the base of the buildings. Thus local ponding levels can be controlled by keeping natural surfaces within the security fence grassed, mowed, and free of high weeds, and by keeping debris from blocking urbanized surfaces. This would prevent water from backing up to higher levels. Ponding on the site is not expected to impact safe operations.

Results for Ponds and Lagoons

To assess whether failures of the local dams could conceivably jeopardize the safety of systems, holding ponds, lagoons, and retention basins formed by these dams were considered in the local drainage analysis. They include the west drainage ditch; X-2230N west-central holding pond, X-2230M southwest holding pond, X-230K south holding pond, east drainage ditch, X-701B holding ponds (northwest, central, and southeast portions), storm sewer L, X-230L north holding pond, X-611B Sludge Lagoon, and X-611A lagoons (north, middle, and south lagoons) (Reference 4). The surface elevations of all but the X-611B are well below the 670-ft minimum grade elevation of the Lead Cascade 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 Lead Cascade facilities. 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 flood wave clearly poses no threat to the Lead Cascade facilities because it could not overtop Perimeter Road (Reference 4).

Results for Ditches and Culverts

The site storm sewer system discharges through each of the outfalls into a series of ditches, culverts, and holding ponds, with eventual discharge to nearby creeks or to the Scioto River directly.

Outfalls at the site have been analyzed to predict their response during a 10,000-year storm (Reference 4). Although some of the culverts would be incapable of carrying the influx of rainwater and some over-banking would happen during a 10,000-year storm, water surface elevations computed for flows in all of the related culverts are below grade elevation at the Lead Cascade facilities and would not cause local flooding at these buildings during a 10,000-year storm.

Effects of Ice and Snow

The PORTS reservation has a generally moderate climate. Winters in the area are moderately cold. On the average, there are 112 days per year below 32^{o}F, but only approximately three days per year at or below 0^{o}F. The average annual snowfall is 22 in. To estimate the extreme snowfall at the site, values for three surrounding cities are used. The maximum monthly snowfalls of record for Columbus (Ohio), Charleston (West Virginia), and Louisville (Kentucky) are 34.4, 39.5, and 28.4 in., respectively, measured in January 1978. If the largest value among the three is used for the site, and if an average density of 0.1 for freshly fallen snow is assumed (Reference 4), this snowfall corresponds to 3.95 in. of rainfall.

1.3.4.3.2 Probable Maximum Flood on Rivers

The maps and the procedure outlined in Section B.3.2.2 of NRC Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants,* were used to estimate the PMF discharge. The log-log plot of the data approximates a straight line. The drainage area of the Scioto River basin above Higby is 5,131 square miles, above Piketon is 5,824 square miles, and above the mouth of the river is 6,517 square miles. The drainage area of the Scioto River above the site (RM 27.5) is estimated from these values to be 6,000 square miles. PMF discharge of the Scioto River at the site as taken from the log-log plot is approximately 1,000,000 cfs. This value is adopted as the PMF discharge near the site (Reference 4).

Coincident Wind Wave Activity

A conservatively high wind velocity of 40 mph blowing over land from the most adverse direction was adopted to associate with the PMF elevation at the site in accordance with Alternatives I and II in Appendix A of NRC Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants*. The fetch length near the site during the PMF of the Scioto River was estimated from USGS topographic quadrangle maps having a 1:24,000 scale to be one mile. The increase of flood elevations of the Scioto River near the site due to this wind wave activity was estimated to be 1.8 ft (Reference 4). The PMF plus this coincident wind wave activity would have a flood stage of 571 ft.

Comparison of Flood Levels with PORTS Reservation Elevations

The nominal, top-of-grade elevation at the site is 670 ft, about 99 ft above the PMF plus wind wave activity flood stage of 571 ft. The top-of-slab floor elevation for the Lead Cascade facilities are 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 (Reference 4). Though the well fields are designated to operate without the effects of flooding, such impacts on the Lead Cascade cooling system would not result in a release of UF_6. Closing strategic valves can isolate the enrichment process, and during severe conditions all or part of the cascade can be shut down.

1.3.4.4 Potential Seismically Induced Dam Failures

The domino-type failure of dams upstream on the Scioto River, failures of individual dams on the tributaries of the Scioto River, and individual dam failures combined with either a 25-year flood or one-half of the PMF of the Scioto River may result in flood elevations that are comparable or even greater than that of the PMF 569 ft. However, even when a conservative wave height of 41.3 ft is used, this cascade of dam failures clearly would not threaten the site because the nominal plant grade elevation is 670 ft, which is 130 ft higher than the normal Scioto River level.

1.3.4.5 Channel Diversions and Ice Formation on the Scioto River

The ancient Newark River was a major channel for alluvium-bearing meltwater from the continental glaciations (Reference 4). This river system ended when its deep valley and those of other major south-draining streams were partially filled with silt, sand, and gravel outwash. The present Scioto River was developed on top of this glacial outwash during the final retreat of glaciers from the area (Reference 4). The Scioto River apparently has a smaller flow and hence a more restricted channel. Therefore, channel diversions of the lower stem of the Scioto River out of the ancient Newark River Valley are unlikely.

Ice occurs on all streams in the Ohio River basin, including its tributary, the Scioto River. Ice on the Scioto River should not affect the water supply to the site because the plant uses groundwater taken from near the river. Additionally, ice formation would not pose a threat of flooding to the site, given the high elevation of the plant relative to the river.

1.3.4.6 Low Water Considerations

Water used at the site can be supplied from wells in the Scioto River alluvium and pumped via existing waterlines to the Water Treatment Plant, X-611. The Pump House, X-608, near the well fields can also pump water from the Scioto River and is a backup system that is used only when the well systems are unable to produce sufficient water to meet the plant demand (Reference 4).

At the Higby gauging station, which is approximately 13 miles north of the PORTS reservation, the minimum river flow measured from 1930 to 1991 was 244 cfs on October 23, 1930 (Reference 4). The consecutive seven-day minimum discharge record of 255 cfs occurred during October 19-25, 1930 (Reference 4). The volumetric river flow is much greater than the site's water use.

1.3.4.7 Dilution of Effluents

The average discharge of the Scioto River near the site is 4,654 cfs. Potentially, this discharge rate has a large capacity for reducing the concentration of received contaminants. For example, the uranium discharged from the site through the local drainage system to the Scioto River was estimated to be 45 kg during 1990 (Reference 4). In 1990, the bulk of the uranium (76 percent) was discharged through Outfall 001 to Little Beaver Creek (Reference 4). Assuming a full dilution, this would result in an average uranium concentration of 1.1×10^{-5} milligrams per liter (mg/L) in the Scioto River.

1.3.5 Subsurface Hydrology

This section describes the subsurface hydrogeologic system in the Interior Low Plateaus region of southern Ohio in the vicinity of the PORTS reservation.

1.3.5.1 Regional and Area Characteristics

In the region surrounding the PORTS reservation in southeastern Ohio, groundwater is used for domestic and municipal drinking water supplies, irrigation, and industrial purposes. Larger demands are usually met by a combination of groundwater and surface water. A system of reservoirs is used for flood control in the Scioto River Basin, which also maintains surface water supplies during periods of low flow.

Aquifers in near-surface sand and gravel deposits adjacent to ancient or present surface drainage courses provide abundant quantities of water. Reliable quantities of groundwater from shallow bedrock aquifers are localized. While abundant quantities of satisfactory groundwater are available from deeper bedrock aquifers, depths as great as 1,000 ft make exploitation of those

aquifers impractical except in the western part of the region. The quality of water from sand and gravel aquifers in the Scioto River Basin is usually classified as fair-to-excellent, while bedrock aquifers are classified as fair because of elevated iron content.

1.3.5.1.1 Aquifers

The subsurface hydrologic system near the site is composed of unconsolidated Pleistocene clastic sediments of glacial and alluvial origin in river valleys and of underlying Paleozoic bedrock units. Figures 1.3-9 and 1.3-10 show the general configuration of these valleys and bedrock units near the site.

The unconsolidated sediments aquifer consists of two distinct aquifers in the immediate vicinity of the site: the Scioto River glacial outwash aquifer and "other" alluvial aquifers, all of Quaternary Age. The Scioto River glacial outwash aquifer consists of permeable deposits of sand and gravel beneath the area adjacent to the river and occupies the ancient Newark River Valley. The other alluvial aquifers consist of deposits of clay and silt interbedded with lenses of sand and gravel, and they partially fill the pre-glacial drainage channels and major tributaries of the Scioto River. These latter aquifers, referred to as the Gallia aquifer of the Teays Formation, are of relatively lesser importance. Because of compositional differences related to their geologic history, the Scioto and Gallia aquifers are treated separately. Table 1.3-4 relates the Scioto River outwash, Gallia hydrogeologic units, and bedrock units to the regional stratigraphic setting.

The bedrock aquifer consists of Silurian through Mississippian limestones, sandstones, and shales. The distribution and use of most of the Silurian and Devonian aquifers is limited to the western portions of the state. For example, groundwater in the Greenfield limestone is used in the area about 50 miles west of the site. The bedrock aquifer near the site consists of the Mississippian-age Bedford Shale, Berea Sandstone, Sunbury Shale, and Cuyahoga Shale in ascending order (Reference 4).

Scioto River Glacial Outwash Aquifer

The Scioto River Valley is underlain by glacial outwash sediments and riverbed alluvium that were deposited during the Quaternary Period. It is one of the principal aquifers in Ohio. The unit extends from the confluence of the Scioto and Ohio rivers to the headwaters of the Scioto in north-central Ohio (Reference 4).

The glacial outwash deposits consist primarily of fine gravel and coarse sand that sometimes is interbedded with fine sand and silt and locally may contain small bodies of clay. These deposits are thickest, 70 to 80 ft, in a comparatively narrow incised bedrock channel, which in the Piketon area, generally underlies the west side of the river valley. The highly porous and permeable glacial outwash deposits are overlain by about 10 to 20 ft of fine-grained, poorly

permeable river alluvium laid down by the modern Scioto River. The water table ranges generally from 10 to 15 ft below the ground surface, and the saturated thickness of the unit is about 40 to 65 ft. For the most part, the aquifer is unconfined (Reference 4).

The Scioto River outwash aquifer supplies municipal, commercial, and domestic water for the area west of the site (Reference 4). The Scioto River outwash aquifer is probably responsive to the stage of the present Scioto River.

Gallia Alluvial Aquifer

The Gallia alluvial aquifer, although similar to the Scioto River outwash aquifer by being Quaternary in age, differs in its geologic history and composition. The Gallia, consisting of silty sand and gravel, is the lower member of the Teays Formation. The overlying Minford Member consists of silt and clay. Where the Sunbury Shale is absent, the Gallia Sand overlies the Berea Sandstone. Because the Gallia represents localized infilling of an ancient streambed, its areal distribution is limited. The Gallia Sand is used locally as a source of water for municipal, commercial, and domestic purposes.

Bedrock Aquifer

Data describing the bedrock aquifer in the region surrounding the site are generally limited to published maps and hydrograph data from the ODNR, Division of Water. Such maps for Pike County and Jackson and Vinton Counties (Reference 4) indicate that the bedrock aquifer serves only domestic needs.

1.3.5.1.2 Regional Groundwater Use

The Scioto glacial outwash aquifer serves as the principal aquifer in the region. Water from this aquifer supplies domestic, agricultural, industrial, and municipal needs. Several municipalities use the aquifer for reserve capacity. Minor alluvial aquifers (including the Gallia) supply domestic needs locally.

1.3.5.1.3 Flow in the Regional Aquifers

Many details of the subsurface hydrologic system have been described at the site in USEC-02 (Reference 4). With respect to aquifer contamination, the two most important aquifers are the Berea Sandstone and the Gallia. The ability for environmental contaminants from facility operations and waste disposal activities to enter these aquifers and migrate off-site is the most important characteristic of the subsurface hydrologic system.

The potential for offsite contamination of regional aquifers is a function of the distribution of geologic units that might enhance cross-formational flow. The vertical head profile between the Berea and the Gallia is determined by the distribution of the Sunbury Shale. Where the Sunbury is absent or very thin, an upward vertical-head profile exists from the Berea to the Gallia. Where the Sunbury is present, a vertically downward head profile exists from the Gallia to the Berea. Thus, the proximity of onsite environmental contaminants to locations

exhibiting downward vertical-head profiles poses the greatest potential for offsite contamination of the Berea. This flow from the Sunbury to the Berea would occur through fractures or deeply weathered zones in the Sunbury.

Groundwater flow at the site is controlled by the complex interactions between the Gallia and Berea units. The flow patterns are also affected by the presence and elevation of storm sewer drainpipes and their bedding and by the reduction in recharge caused by building and paved areas. Three principal discharge areas exist for all ground water: (1) Little Beaver Creek to the north and east; (2) Big Run Creek to the south; and (3) two unnamed drainages to the west. Groundwater flow patterns in both the Berea and Gallia are characterized by an east-west trending groundwater divide that passes through the site. Other groundwater divides are also present, dividing the flow system of each unit into four sub-basins in the Gallia and three in the Berea.

While contamination of the Berea aquifer from onsite activities is possible, due to the downward vertical-head profile from the Gallia, offsite monitoring has not detected contaminant concentrations above background levels (Reference 4). Additionally, dissolved solids exceeding 10,000 ppm within about five miles down gradient from the site make it unlikely that significant portions of the Berea drinking water resource would be adversely affected.

Precipitation is the primary source of recharge of these aquifers. Recharge at the site is estimated at between 2.3 and 11.7 in./year (Reference 4). Infiltration reaches the water table and moves laterally to areas of discharge or vertically to adjacent aquifers. The Gallia aquifer near or adjacent to surface drainage ways is likely in active communication with the surface water.

1.3.5.2 Site Characteristics

The site sits in a mile-wide former river valley (Portsmouth River Valley) surrounded by farmland and wooded hills with generally less than 100 ft of relief. The main plant area has a nominal elevation of 670 ft amsl about 130 ft above the stage of the Scioto River, which lies about 2.5 miles to the west. The Scioto River and its tributaries receive essentially all of the surface water and groundwater discharge at the site.

Geologic units controlling groundwater flow beneath the site are, in descending order, the Minford and Gallia unconsolidated units of the Quaternary age, and the Sunbury, Berea, and Bedford bedrock units of the Mississippian age (Table 1.3-4). The Mississippian Cuyahoga shale, the youngest bedrock unit in the area, forms the hills east and west of the main plant site. Also present in some places is up to 20 ft of artificial fill, which is predominantly Minford silt and clay.

The main groundwater flow system beneath the site is the Gallia sand and the lower unit of the Minford, the Minford silt. The Gallia sand and the lower Minford silt form the uppermost, unconfined aquifer (the Gallia aquifer) with a combined thickness of about 11 ft (Figure 1.3-11). The bottom of the Gallia aquifer has an elevation ranging from 630 to 640 ft amsl in the plant area.

The Gallia aquifer is partly surrounded by the Cuyahoga shale, which lies in the wooded hills around the site. The Sunbury shale underlies both the Gallia aquifer and the Cuyahoga shale. The Sunbury separates the Gallia aquifer from the underlying confined aquifer, the Berea sandstone. Where the Sunbury is absent or thin, the Berea aquifer and the overlying Gallia aquifer act essentially as one unit. About 100 ft of Bedford shale underlies the Berea aquifer over the entire site. The lower 10 ft of the Berea is very similar to the underlying Bedford shale (Reference 4).

1.3.5.2.1 Aquifers Beneath the Site

The Gallia exhibits the highest hydraulic conductivity of all aquifers on the site. Hydraulic conductivity values range from 0.11 to 150 feet per day (ft/d), with a mean of 3.4 ft/d (Reference 4). Groundwater flow directions in the Gallia are roughly from the center of the site toward the surrounding low-lying surface water drainage system. The ultimate discharge area for most groundwater is Little Beaver Creek to the north and east, Big Run Creek to the south, and two unnamed drainages to the west.

1.3.5.2.2 Aquifer Properties

At the site, the Berea Sandstone exhibits little spatial variation in hydraulic properties. The site-wide mean hydraulic conductivity for the Berea is 0.16 ft/d (Reference 4). The highest hydraulic conductivity in the Berea was measured as 0.35 ft/d at the X-616 area, where the unit has been slightly eroded and may be slightly weathered; the lowest hydraulic conductivity was measured is 0.1 ft/d at both X-231B and X-701B.

Groundwater elevations in the Berea Sandstone are determined by local geologic conditions. Measurements at the site between August 1988 and September 1989 indicate a mean water elevation of 646.15 ft mean sea level (MSL) with a standard deviation of 0.92 ft (Reference 4). A generally downward vertical gradient occurs between the Berea and overlying aquifer when overlain by the Sunbury Shale, which acts as an effective confining unit. Where the Sunbury is absent or very thin, an upward vertical gradient exists between the Berea and overlying aquifer. Groundwater flow in the Berea is expected to be similar to those of the Gallia except in the eastern part of the site, where the directions are generally toward the east and southeast.

Recharge from precipitation has been estimated to be 8.9 in./year using the 1985 data and the Thornthwaite method (Reference 4). This corresponds to about 25 percent of the total precipitation of 35.78 in. that year. In general, the estimated annual recharge rates vary from 3.3 to 11.7 in./year.

Little Beaver Creek to the north and east, Big Run Creek to the southeast, and the two unnamed tributaries to the west control groundwater flow in the Gallia and Berea aquifers by acting as local recharge or discharge areas. In some places, the large-diameter storm drain segments are partially below the elevation of the Gallia water table (Reference 4). These drains and surrounding gravel beddings may act as groundwater interceptors in the Gallia flow system.

1.3.5.2.3 Groundwater Flow

The main groundwater flow unit beneath the PORTS reservation is the Gallia aquifer formed by the Gallia sand and the Minford silt, with a combined average thickness of about 11 ft. The hydraulic conductivity of this aquifer is not considered as high, but the surrounding Cuyahoga shale and underlying Sunbury shale and Berea sandstone have even lower conductivities and form less important groundwater flow units (Reference 4). In general, the Gallia aquifer beneath the main plant area receives recharge through infiltration of rainfall and discharges water to surrounding low-lying areas through openings formed by missing Cuyahoga shale. One narrow opening is between the X-701B area and Little Beaver Creek to the east. Two wide openings exist, one near the northern perimeter road toward Little Beaver Creek and the other near the southern perimeter road. Discharges, in the form of groundwater, are likely to occur from the main plant area through these openings. Other openings that are not easily seen from the bedrock surface plot are associated with Big Run Creek to the south and the two unnamed tributaries to the west. Discharges through these openings are likely first in the form of groundwater and then as surface water in the creeks. All these discharge routes can be potential pathways for the site contaminants to reach areas outside the plant and ultimately the Scioto River.

Regional flow in the Berea is generally to the southeast, in the direction of structural dip. Locally, the flow direction is affected by Big Run Creek, Little Beaver Creek, and the west and southwest drainages (Reference 4). For example, flow in the northern part of the site turns somewhat northward due to the influence of Little Beaver Creek. In areas where the Sunbury is absent, the Berea and the overlying Gallia become hydraulically connected.

Groundwater flow directions in both aquifers are influenced by the presence of Little Beaver Creek, Big Run Creek, and the two unnamed tributaries. At many places, the two groundwater flow systems are roughly parallel, but at some places, for example near the northern perimeter road, they are quite different. In general, large head differences exist between the Gallia and the Berea because the Sunbury shale presents an effective barrier that restricts the vertical communication between the two aquifers (Reference 4).

Offsite monitoring of the sanitary water systems of local residents near the site began in 1979, and analysis for the presence of organic compounds was added in 1986 (Reference 4). The monitoring is conducted semiannually on springs and private wells near the site including parameters such as uranium, technetium, total alpha, and total beta. To date, the monitored parameters have not been detected above background levels in any of the sampling locations.

1.3.6 Geology and Seismology

1.3.6.1 Regional and Site Physiography

The PORTS reservation is located within the Interior Low Plateaus physiographic province, about 20 miles south of its northwestern edge. It is bordered on the north and west by the Central Lowlands province and on the south and east by the Appalachian Plateaus province. The Interior Low province is underlain by relatively flat-lying Paleozoic Age limestone and shale.

Portions of the Interior Low Plateaus province have been glaciated, but the site is south of the region covered by Pleistocene glaciation. However, alluvium and transported glacial sediments form a surface veneer in the mile-wide, broad valley where the site is located. The surrounding hills have been maturely dissected by erosion, exposing the underlying, nearly flat-lying shale and sandstone of Mississippian and Pennsylvanian Age.

The plant is located within a broad, flat valley that was (1) primarily developed by long-term erosion of the shale and sandstone that underlies the Interior Low Plateaus physiographic province; (2) subsequently modified by partial filling by glacial and alluvial sediments; and (3) later subjected to erosion. The prolonged erosion since the Permian Period has produced the dominant topography. Ground elevations within the plant generally range from about 660 ft MSL to 680 ft MSL, although the ground rises to about 700 ft MSL at the base of hills that border the Perimeter Road; the surrounding hills extend up to about 1,200 ft MSL. The nearby Scioto River (at about elevation 510 ft MSL) is the lowest elevation within five miles.

Prior to construction of the GDP, the area was farmland that formed a portion of the watershed for the nearby Scioto River. A drainage divide (about elevation 675 ft MSL) was at approximately midpoint of the plant, which separated gullies and streams flowing to the north from those flowing west and south. Generally, site preparation and grading performed approximately 50 years ago involved only minor surface modification. With the exception of a few drainage features (swales) that required as much as 20 ft of fill, most of the area developed was cut less than 10 ft and filled less than 12 ft.

1.3.6.2 Site Geology

Aside from roadways and other ancillary structures outside the Perimeter Road, the plant is located within the valley eroded into the bedrock by the ancient Portsmouth River and later filled in by glacial lake sediments. Except for a few low hills that extend into the plant site, the Perimeter Road on the west and east generally follows the lateral limits of the ancient Portsmouth River Valley. The valley is bounded on the west by a series of low hills extending up to elevation 840 ft MSL that have been maturely dissected; these hills expose nearly flat-lying Mississippian Age shales of the Sunbury and Cuyahoga Formations. The Sunbury and Cuyahoga Formations are also exposed in the maturely dissected low hills east of the plant site. These consolidated Mississippian formations dip downward to the east about 27 ft/mile (i.e., less than ½ a degree).

Drainage that developed at the site prior to glaciation consisted of a northward and westward flowing master stream (the ancient Teays River) and tributaries such as the ancient Portsmouth River. The Portsmouth River deposited a thin discontinuous veneer of alluvium in the site valley that has subsequently been covered by lacustrine deposits of glacial origin. Only the small streams that flow through the site contain recent alluvium.

Unconsolidated deposits at the site consist of Quaternary stream alluvium (Holocene and Pleistocene), Pleistocene lacustrine deposits of glacial origin, and older alluvium of the ancient Portsmouth River. Consolidated deposits within 500 ft of the ground surface consist of Devonian, Mississippian, and Pennsylvania shale and sandstone.

Unconsolidated material

Fill – Fill was placed during the 1950s to develop the site. Most of the fill ranges from 1 ft to 3 ft in thickness, but up to 20 ft of fill was placed in former stream valleys or draws to develop a plateau for building construction for the GDP facilities. Then in the early 1980s, additional fill was placed to create plateaus for the GCEP building construction. The fill is composed mostly of clean, silty clay. Verification data regarding fill density and its moisture content indicate that the fill under the plant buildings was compacted to at least 95 percent of its maximum dry density according to ASTM D 698 (standard Proctor).

Lacustrine deposits – Lacustrine deposits averaging 23 ft in thickness are exposed at the ground surface over much of the site and underlie fill at the remainder of the site; these deposits have been termed the Minford clays, Minford silts, or the Minford Clay Member of the Teays Formation. The general soil profile is composed of about 16 ft of clay underlain by about 7 ft of silt. Both these soil types are firm to very stiff, overconsolidated, and classified as silty clay and silt, but some highly plastic clay occurs near the ground surface.

Older alluvium – The lacustrine deposits are underlain by a discontinuous interval of clayey sand and gravel (Gallia sand) deposited by the ancient Portsmouth River. The alluvium is commonly referred to as the Gallia Sand Member of the Teays Foundation in the nearby Teays Valley. The average thickness is about 3 ft; the maximum thickness of the alluvium is 12 ft. It is firm to dense.

Consolidated material

Cuyahoga Formation – This Mississippian formation crops out in hills adjacent to the site, with the base of the formation at elevation 639 ft MSL. When unweathered, the Cuyahoga consists of about 339 ft thickness of hard grey to grey-green shale with lenses of sandstone.

Sunbury Formation – Underlying the Cuyahoga is a 19 to 20 ft thick interval of hard, black, carbonaceous shale. It underlies the unconsolidated sediments beneath most of the plant site.

Berea Formation – The Berea Formation underlies the Sunbury shale and extends downward. It is composed of about 30 to 35 ft of grey thick-bedded, fine-grained sandstone with shale laminations.

Bedford Formation – The Bedford is composed of about 98 ft of varicolored shale with interbeds of sandstone and siltstone.

Ohio Formation – The Ohio Shale is the uppermost Devonian Formation under the plant site. It is composed of 300 to 600 ft of dark brown, dark grey, and black fissile shale.

1.3.6.3 Site Structural Setting

Lacustrine deposits cover essentially all of the site bedrock; some streambeds contain recent alluvium. Little bedrock is exposed at the site except in the hills surrounding the plant. Neither the U. S. Army Corps of Engineers studies nor the Law Engineering Study in 1978 discovered evidence of bedrock faulting. The available data indicates that the underlying bedrock is not faulted; it has a strike of north 28^{o} east and a homoclinical dip to the southeast of about 1/2 a degree.

1.3.6.4 Engineering Geology

The available evidence indicates the favorable performance of the PORTS facilities since their construction in the 1950s and the more recent GCEP facilities constructed in the early 1980s with respect to bearing capacity, settlement, and modest seismic events.

No shears, folds, or other structural weaknesses are known to be in the bedrock. Measurements of joint sets in bedrock exposed around plant site exhibit jointing typical of undeformed bedrock. These joints have no effect on the performance of foundations since they are covered by an interval of lacustrine glacial deposits. No evidence from the borings indicates zones of deep weathering that might indicate faulting or shearing.

No published data exist on unrelieved stresses in the bedrock, but the geologic history suggests that the bedrock may still be undergoing a very slow isostatic rebound. This rebound is due to a combination of the past loading and subsequent unloading of the bedrock by the Pleistocene glaciers and/or stress relief from erosion of the unconsolidated lacustrine sediments.

The consolidated bedrock within 500 ft of the ground surface is predominately clastic in origin (shale and sandstone).

Most of the unconsolidated soils are cohesive and overconsolidated and relatively uniform in thickness and extent. The soils exhibit a low potential for liquefaction and differential settlement. Cohesive soils exposed at the surface may exhibit minor shrinkage cracks resulting from moisture loss.

The geologic literature and records of mineral production in the site area indicate no mineral extraction has been done beneath the site. The potential exists for minor oil and gas accumulations in the underlying consolidated strata, but there are no records of significant gas or oil production within five miles of the site.

The soil at the site is primarily low plasticity clay and silty clay. The bedrock is composed of hard shale and sandstone.

The regional geologic history and extensive amount of exploratory data indicate no evidence of tectonic depressions, shears, faults, or folds.

The plant uses process water from the aquifer below the Scioto River, and no groundwater is withdrawn from the subsurface at the plant site for sanitary or process uses.

The exploratory and laboratory test data indicate that the glacial and alluvial soils are overconsolidated and have moisture contents well below their liquid limit. Engineering studies have shown the soils are only moderately compressible under applied foundation loads, and the satisfactory performance of the various foundations attests to that. The potential is low for surface fissuring of soils resulting from a period of extreme drought.

The studies by the U. S. Army Corps of Engineers and Law Engineering in the 1970s in the GCEP area, south-southeast and southwest of the GDP, found groundwater between 650 ft MSL and 665 ft MSL. The basal older alluvium exhibits no evidence of artesian conditions. Limited data on groundwater fluctuations indicate variations of between 3 ft and 5 ft over a period of six months. The groundwater level responds to annual precipitation.

No problems were encountered with groundwater during construction of the GCEP facilities. Most foundations bear upon the stiff lacustrine soils at depths of 5 ft or less below the finished floor elevation of the buildings.

No slopes within the Perimeter Road have inclination of 3 horizontal (H):1 vertical (V) or greater except for one slope; this slope is not adjacent to any structures (Reference 4). Low inclination slopes less than 20 ft in height that have soil parameters of $\phi = 10^{o}$, c = 1,000 will have a static safety factor of at least 2.0 and a dynamic safety factor of at least 1.5 under a peak ground acceleration (PGA) of 0.21 gravity. The natural ground and engineered fill upon which the structures are founded have been analyzed for shear failure and settlement. Design documents show the factor of safety against shear failure under static conditions is more than 2.0, and predicted total settlements of foundations are less than 2 in. Because of the stiff nature of the foundation soils, negligible settlement will occur as a result of the design basis earthquake, as discussed in the next section.

1.3.6.5 Seismology

There are no major geologic fault structures in the vicinity of the site and there have been no historical earthquake epicenters within less than 25 miles from the site. However, there have been eight earthquake epicenters within 50 miles. The maximum event had an epicenter intensity of over IV on the Modified Mercalli (MM) scale. But all of these events were at the site with intensities between I and IV. The maximum PGA of a MM level IV event roughly corresponds to 0.02 gravity. Historically, the maximum earthquake-induced PGA experienced at the site was in 1955 and had a value of only 0.005 gravity.

In the Preliminary Safety Analysis Report developed for GCEP during the 1980s, the DOE documented the results of studies of the historic seismicity of the area surrounding the PORTS reservation. Data was developed on probable seismic activity and the intensity levels were converted into acceleration values. The maximum earthquake was defined as one with a mean recurrence interval of 1,000 years. This corresponds to an earthquake with a horizontal PGA of 0.15 gravity. Thus, the DOE considered that it was sufficient to design the structures, systems, and components necessary for safety to withstand this level earthquake without leading to undue risk to the health and safety of workers, the public or the environment. That is, the 1,000-year return earthquake was the design basis earthquake (DBE) for GCEP.

Several studies, including those mentioned above, have been conducted specifically for determining the seismic hazard for the GCEP site. One such study conducted by Beavers was involved in establishing the seismic design criteria for the GCEP. This criteria was published in a DOE document, ORO-EP-120 in 1978 and contained recommended design and maximum earthquake PGA values to be used in the design. The PGA values corresponding to these two earthquake levels were 0.04 gravity for the design earthquake and 0.l5 gravity for the maximum earthquake corresponding to 72- and 1,000-year return periods, respectively. These PGA levels were selected based on judgement considering: (l) much of the information discussed in other former studies of the GDP site; (2) the GCEP was to be a newly constructed facility; (3) the GCEP might be subjected to licensing requirements; and (4) return periods of 1,000 years for events concerning safety were discussed for new enrichment plants. Although recommended, it was the opinion of the authors of ORO-EP-120 that the PGA value of 0.15 gravity for a return period of 1,000 years was on the conservative side. Therefore, the DBE for the Lead Cascade is the 1,000-year return earthquake.

1.3.6.6 Surface Faulting

The geologic setting of the site suggests there is a low probability of faulting within five miles of the site. No data from the three extensive geotechnical studies at the site (rock shearing, sharp changes in strata dip, and flexures) are characteristic of faulted rocks. The available data indicates the site bedrock is not faulted.

1.3.6.7 Liquefaction Potential

Three extensive exploration and laboratory testing programs (data sets) have been completed at the site, with the total number of approximately 960 exploratory borings. These borings and accompanying laboratory test results were used at the site to analyze the response of soil to ground shaking caused by earthquakes.

The laboratory classification tests, shear strength tests, and consolidation test data were used to define the general engineering characteristics of the soil. Analysis of the data indicates that there is a low potential for soil liquefaction at the site, even in the unlikely event of the occurrence of an earthquake of magnitude 5.25 with a maximum PGA of 0.15 gravity. Consequently, settlement in the site area due to liquefaction is unlikely.

Table 1.3-1
Historic and Projected Population in the Vicinity of the PORTS Reservation

	1980	**1990**	**2000**	**2010**
Jackson County	30,592	30,230	32,641	34,724
Pike County	22,802	24,249	27,695	29,981
Ross County	65,004	69,330	73,345	80,111
Scioto County	84,545	80,327	79,195	81,307
Region of Influence	202,943	204,136	212,876	226,123
Ohio	10,797,630	10,847,115	11,353,140	11,805,877

Year 2010 projections based on established rates applied to 2000 census counts.

(Reference 2)

Table 1.3-2
Precipitation as a Function of Recurrence Interval and Storm Duration for the PORTS Reservation

Recurrence Interval (years)	Storm duration (hours) 0.5	1	2	3	6	12	24
	Precipitation (in.[a])						
1	0.85	1.06	1.34	1.44	1.75	2.04	2.43
2	1.04	1.28	1.57	1.71	2.02	2.44	2.70
5	1.36	1.66	1.98	2.14	2.52	2.98	3.41
10	1.52	1.93	2.30	2.52	2.98	3.40	3.90
25	1.75	2.24	2.64	2.92	3.38	3.91	4.55
50	1.96	2.51	2.97	3.16	3.78	4.20	4.93
100	2.16	2.73	3.22	3.48	4.00	4.88	5.26
10,000[b]	3.46	4.45	5.15	5.57	6.42	7.49	8.32

a. 1 in. = 2.54 centimeters (cm)
b. Extrapolated values calculated using least-squares methodology.

(Reference 4)

Table 1.3-3
Comparison of Flood Elevations of the Scioto River near the PORTS Reservation with the Nominal Grade Elevation

	Elevation	
Recurrence interval	Meters (m)	Feet (ft)
50-year flood[a]	170.1	558.0
100-year flood[a]	170.8	560.3
500-year flood[a]	172.4	565.7
Historical written record[b]	169.7	556.7
Probable Maximum Flood[c]	174.0	571.0
PORTS nominal grade	204.2	670.0

a. Estimates by U.S. Army Corps of Engineers (Reference 4).
b. Estimated from records at Higby, 181.0 m (593.7 ft) (Reference 4), assuming the flood level at the mouth of Big Beaver Creek is 11.3 m (37 ft) lower.
c. Probable Maximum Flood (PMF) calculated flow is greater than that of the estimated 10,000-year flood discharge.

(Reference 4)

Table 1.3-4
Regional Stratigraphic and Hydrogeologic Subdivisions

ERA	System	Series	Formation or Unit	Hydrogeologic Unit
Cenozoic	Quaternary	Pleistocene	Teays Scioto River Outwash Minford Member Gallia Member	Scioto River
	Mississippian		Cuyahoga Sunbury Shale Berea Sandstone Bedford Shale	Gallia
Paleozoic	Devonian	Upper	Ohio Shale	Bedrock

(Reference 4)

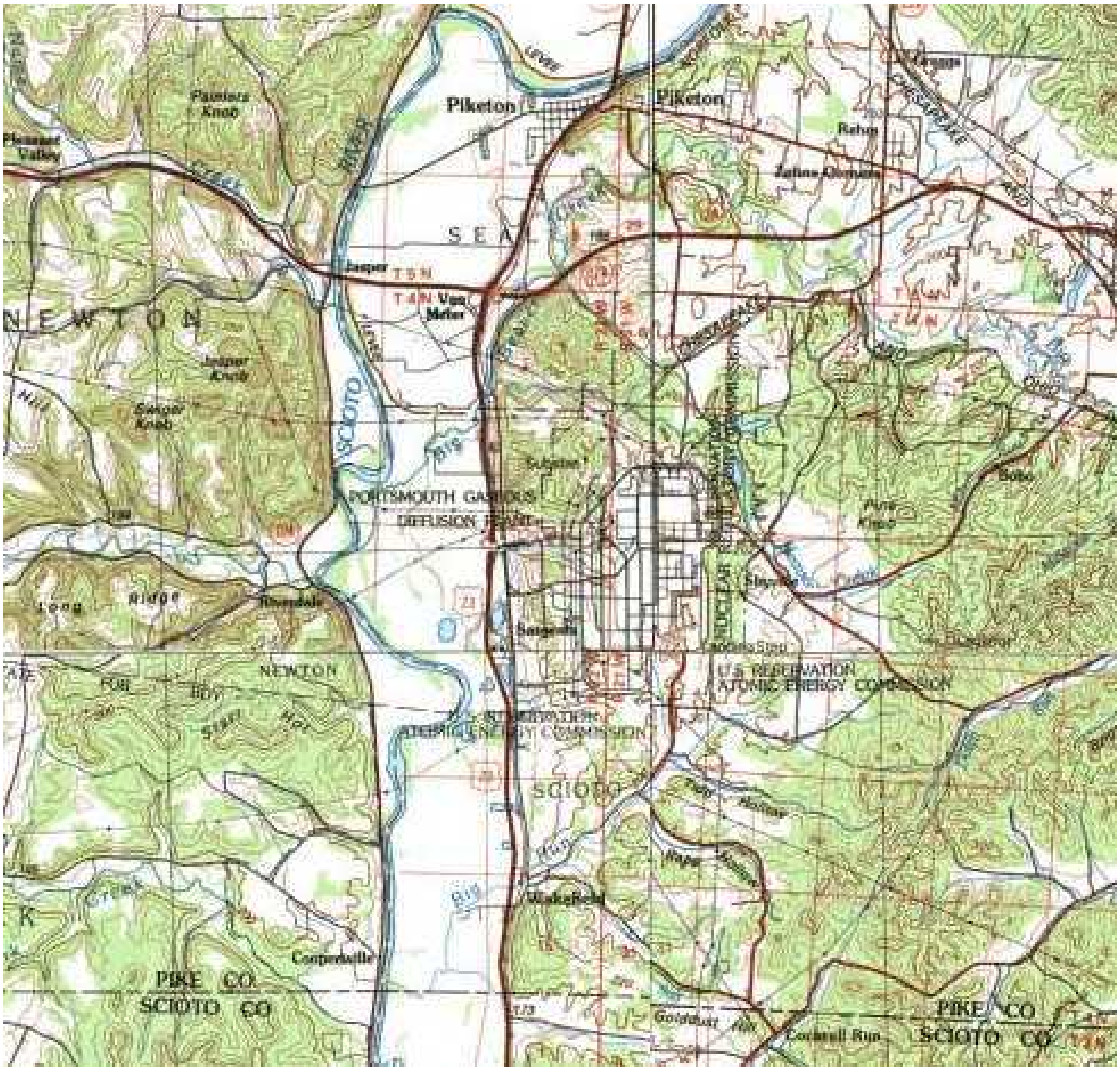

Figure 1.3-1
Topographic Map of the PORTS Site
(Reference 6)

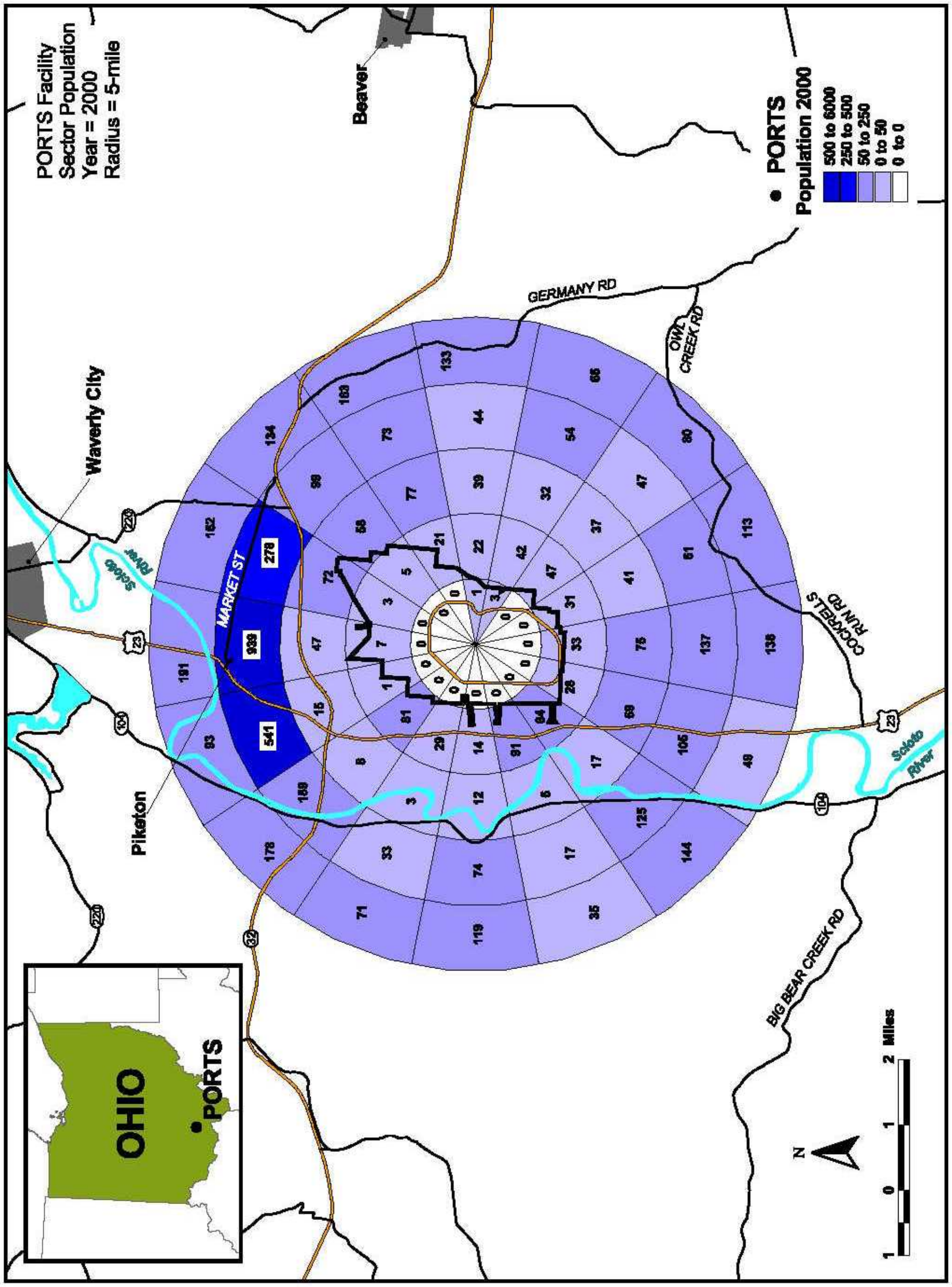

Figure 1.3-2
Population within 5-Mile Radius of PORTS
(Reference 7)

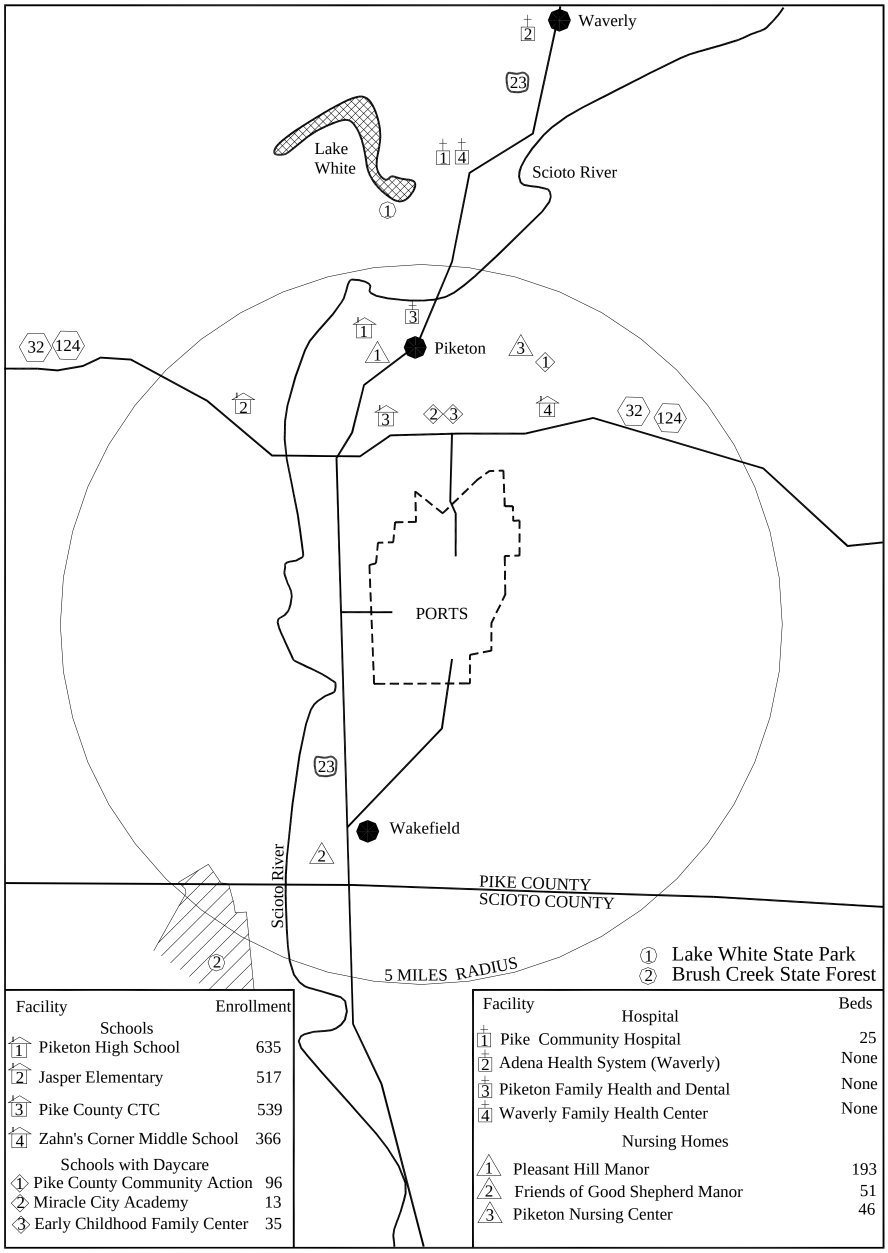

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Figure 1.3-3
Special Population Centers within 5 miles of PORTS

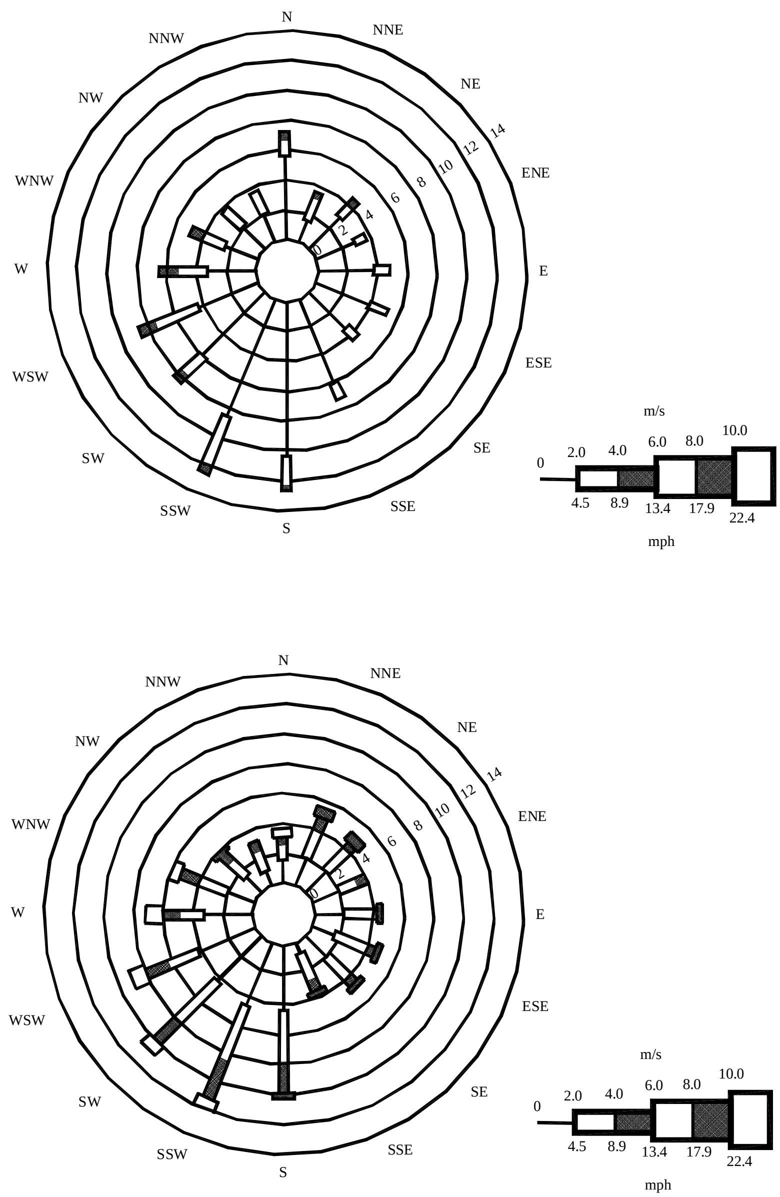

Figure 1.3-4
Comparison of wind roses at 10-m (top) and 32-m (bottom) levels at PORTS for 1993
(Reference 4)

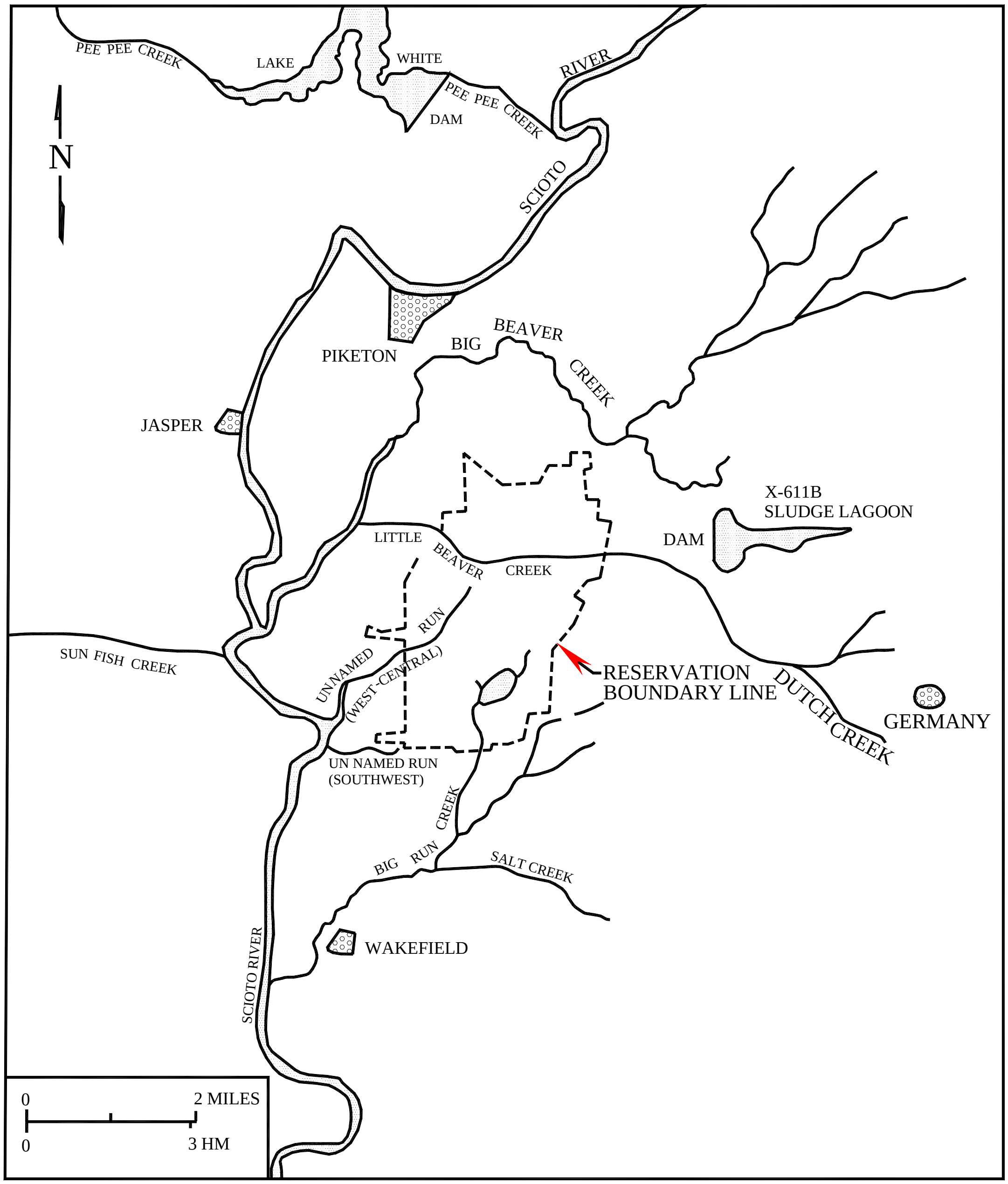

Figure 1.3-5
Location of Rivers and Creeks in the Vicinity of PORTS
(Reference 4)

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 1.3-6
Ponds and Lagoons at PORTS
(Reference 4)

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 1.3-7
Elevations of Roadways and of the Surrounding Areas of Main Process Buildings
(Reference 4)

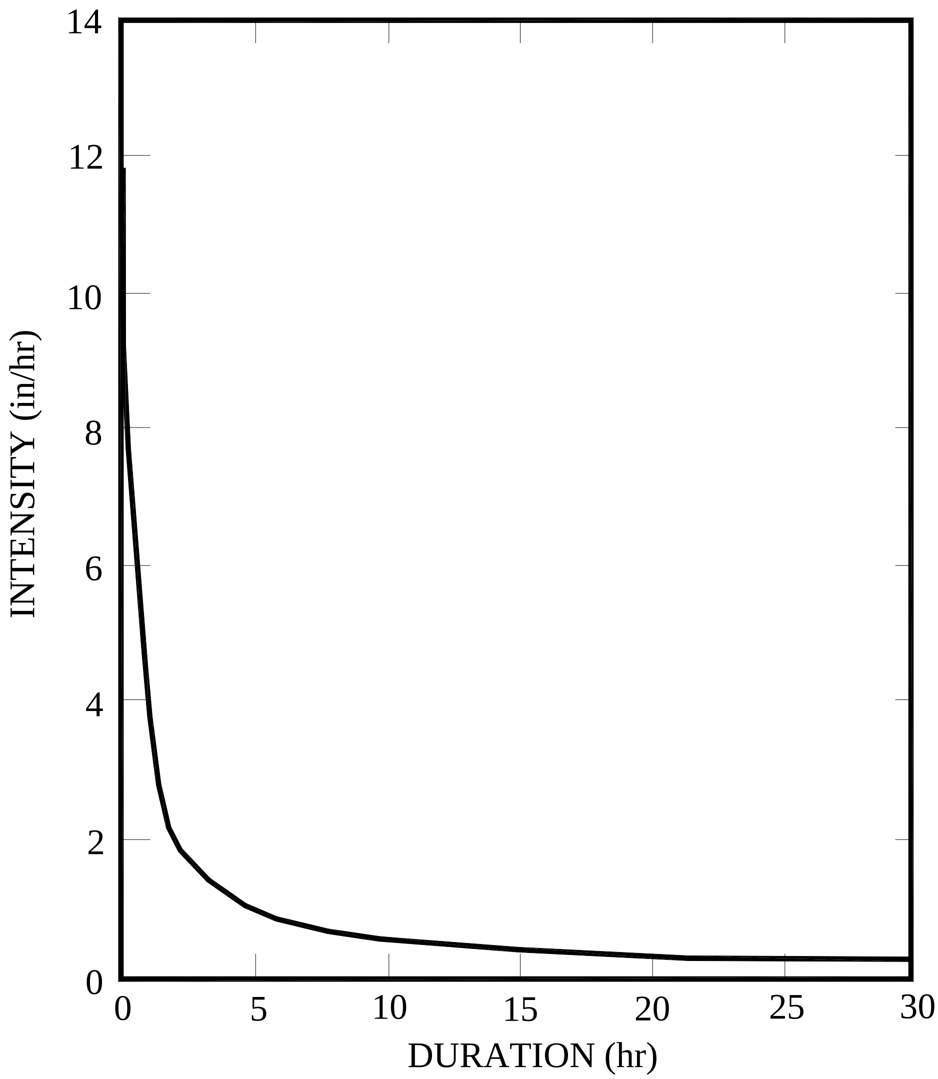

Figure 1.3-8
The 10,000-year Intensity Versus Duration Graph for PORTS
(Reference 4)

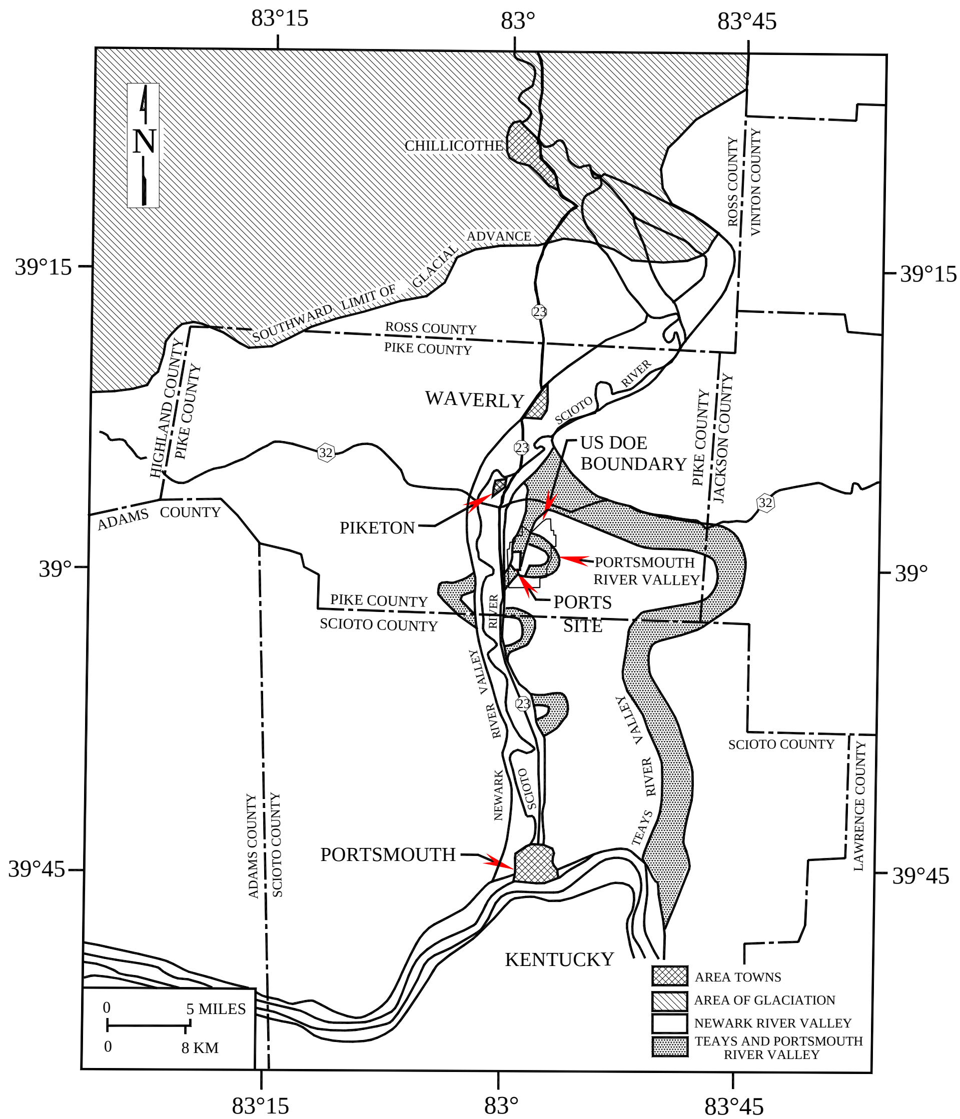

Figure 1.3-9
Location of the Ancient Newark (Modern Scioto) and Teays Valleys in the PORTS Vicinity
(Reference 4)

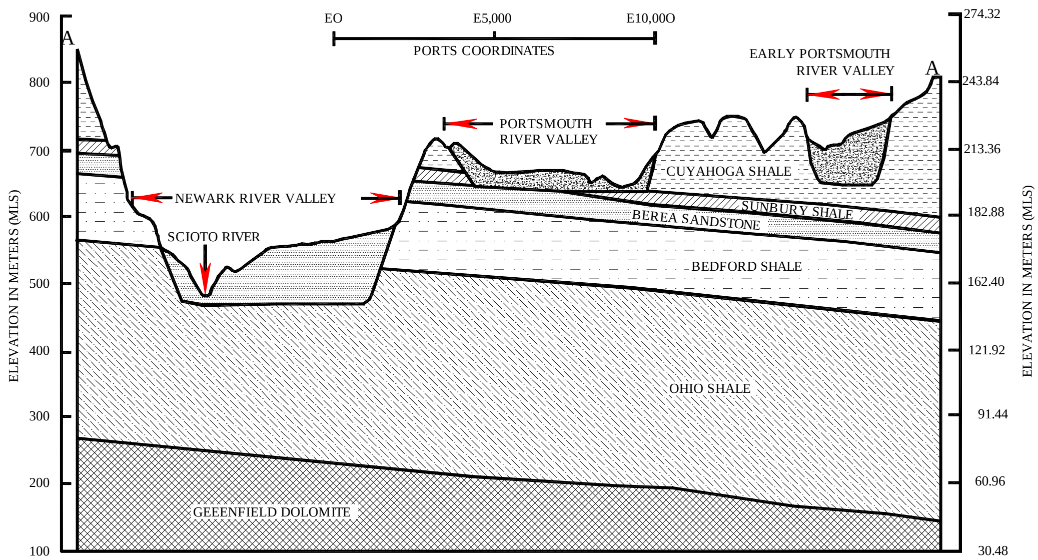

Figure 1.3-10
Geologic Cross Section in the PORTS Vicinity
(Reference 4)

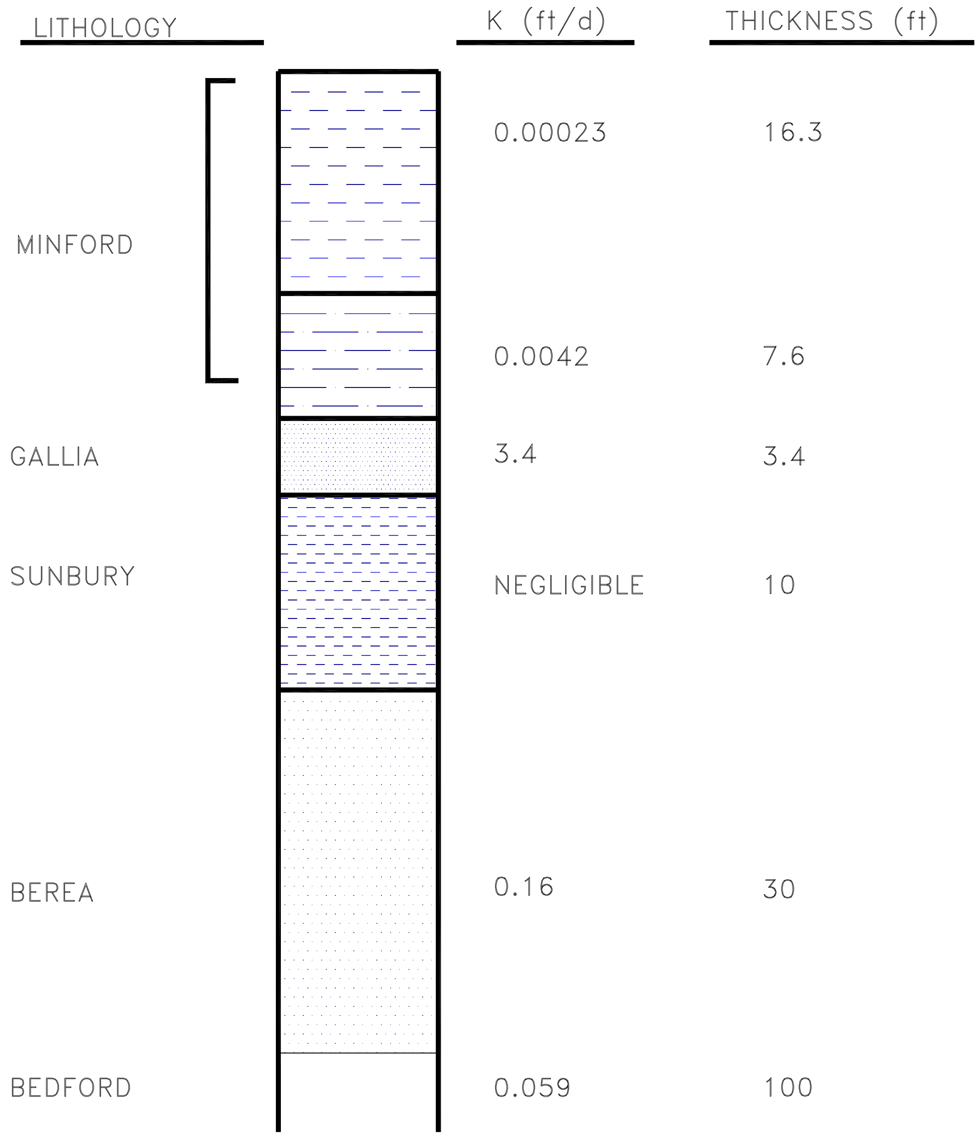

Figure 1.3-11
Geologic Column at PORTS
(Reference 4)

1.4 Applicable Codes, Standards, and Regulatory Guidance

The following lists the various industry codes, standards, and regulatory guidance documents that have been referenced in this license application. The extent to which the Licensee satisfies each code, standard, and guidance document is identified below:

1.4.1 American National Standards Institute/American National Society

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

  The Licensee 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*

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

  For the reference to this standard see Section 11.4.4.1 of this license application.

- ANSI/ANS-8.1-1983, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactor*

  The Licensee satisfies the guidance of this standard with the following exceptions/clarification:

  > Section 4.1.6 - Operations are reviewed annually; however, this review is performed by personnel in the operating group who are knowledgeable of the NCS requirements for their operations. Personnel who are knowledgeable in NCS and are independent of operations (e.g., Engineering) provide assistance in these annual reviews. Personnel who are knowledgeable in NCS and are independent of operations (e.g., Engineering) review operations biannually (every two years).

  For references to this standard see Sections 5.4.1, 5.4.2, 5.4.5.1, and 5.4.5.2 of this license application.

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

  The Licensee satisfies the provisions of this standard with the following exceptions/clarification:

Section 7.8 - Operations are reviewed annually; however, this review is performed by personnel in the operating group who are knowledgeable of the NCS requirements for their operations. Personnel who are knowledgeable in NCS and are independent of operations (e.g., Engineering) provide assistance in these annual reviews. Personnel who are knowledgeable in NCS and are independent of operations (e.g., Engineering) review operations biannually (every two years).

For references to this standard see Sections 5.4.1, and 11.3.12, of this license application.

- ANSI/ANS-8.20-1991, *American National Standard for Nuclear Criticality Safety Training*

  The Licensee satisfies the provisions of this standard.

  For references to this standard see Sections 11.3.4.2, 11.3.6, and 11.3.11 of this license application.

1.4.2 American National Standards Institute

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

  The Licensee utilizes the provisions contained in Section 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*

  The Licensee 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, *American National Standard for Nuclear Materials – Uranium Hexafluoride – Packaging for Transport*, 2001.

  The Licensee commits to meet the guidelines of ANSI N14.1 with the following exemptions/clarifications:

  A. Sections 6.9.1(2) and 6.9.7(3) and Figure 13a – The minimum design metal temperature (MDMT) of nickel or Monel body UF_6 cylinders is -325°F, rather than -40°F, during operations described in Section 1.1.2.5.4 of this license application.

B. Figures 6 and 13 – The nameplate can be attached with welds approximately 1 inch around each corner, rather than continuous welding.

For reference to this standard, see Section 7.3, of the Integrated Safety Analysis Summary.

1.4.3 American National Standards Institute/American Society of Mechanical Engineers

- ANSI/ASME NQA-1-1994, *Quality Assurance Requirements for Nuclear Facility Applications*

  The Licensee satisfies the provisions of this standard as stated below, with clarification stated in the QAPD:

  A. The Licensee satisfies the definitions, as stated in the Introduction of Part I of ASME NQA-1-1994.

  B. Indoctrination and training satisfies the provisions of Supplement 2S-4, “Supplementary Requirements for Personnel Indoctrination and Training” of Part 1 of ASME NQA-1-1994.

  C. Quality Control personnel performing inspection and testing satisfies the provisions of Supplement 2S-1, “Supplementary Requirements for the Qualification of Inspection and Test Personnel” of Part 1 of ASME NQA-1-1994.

  D. QA audit personnel satisfies the provisions of Supplement 2S-3, “Supplementary Requirements for the Qualification of Quality Assurance Program Audit Personnel” of Part 1 of ASME NQA-1-1994.

  E. Design outputs that consist of computer programs are developed, validated, and managed in accordance with ASME NQA-1-1994, Basic Requirement 11 and NQA-1, Part II, Subpart 2.7, “Quality Assurance Requirements for Computer Software for Nuclear Facility Application.”

  F. Methods of design verification satisfies the provisions of Supplement 3S-1 of ASME NQA-1-1994.

  G. Computer Program Testing is performed in accordance with ASME NQA-1-1994, Basic Requirement 11, “Test Control,” and Supplement 11S-2, “Supplementary Requirements for Computer Program Testing.”

  H. Lifetime records are defined in accordance with ASME NQA-1-1994, Supplement 17S-1, “Supplementary Requirements for Quality Assurance Records,” Section 2.7.1.

I. Hard copy or microfilm storage facilities satisfies the guidance of ASME NQA-1-1994, Supplement 17S-1, "Supplementary Requirements for Quality Assurance Records," Section 4.4.

For the references to this standard see Section 11.5.1 of this license application and Sections 2, 3, and 11 of the QAPD.

1.4.4 American Society of Mechanical Engineers

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

  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 QAPD

  Appendix A - Do not sample adsorbents

  Appendix B - Do not use allowable leakage guidance

  Appendix C – This appendix is used as guidance only

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

  For the reference to this standard see Section 4.6.1 of this license application.

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

  New and existing fixed HEPA filter systems that satisfy the requirements of ASME N509 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:

  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

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

Existing fixed HEPA filter systems that do not satisfy the requirements of ASME N509 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 this license application.

1.4.5 National Fire Protection Association

- NFPA 10-1990, *Standard for Portable Fire Extinguishers*

  The Licensee satisfies the provisions of this standard with the following exceptions/clarification:

  > The provisions of this standard were used as guidance only in determining the size, selection, and distribution of portable fire extinguishers. The Licensee will satisfy the provisions of this standard for modifications to the facility except as documented and justified by the Authority Having Jurisdiction (AHJ).

  For references to this standard see Sections 7.1.1, and 7.5.3 of this license application.

- NFPA 13-1989, *Standard for the Installation of Sprinkler Systems*

  The Licensee satisfies the provisions of this standard with the following exceptions/clarification:

  > The provisions of this standard were used as guidance only for the design and installation of wet and dry pipe automatic sprinkler systems. In addition, the Process Building meets the definition of Ordinary Hazard Occupancies (Group 1) as stated in this standard and the fire protection system is designed and installed to meet or exceed the requirements associated with Ordinary Hazard Group 1. The Licensee will satisfy the provisions of this standard for modifications to the facility except as documented and justified by the AHJ.

  For the reference to this standard see Section 7.1.1 of this license application.

- NFPA 15-1990, *Standard for Water Spray Fixed Systems for Fire Protection*

  The Licensee will satisfy the provisions of this standard for modifications to the facility except as documented and justified by the AHJ.

  For the reference to this standard see Section 7.1.1 of this license application.

- NFPA 17-2002, *Standard for Dry Chemical Extinguishing Systems*

  The Licensee will satisfy the provisions of this standard for modifications to the facility except as documented and justified by the AHJ.

For the reference to this standard see Section 7.5.3 of this license application.

- NFPA 24-1992, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*

  The Licensee satisfies the provisions of this standard with the following exceptions/clarification:

  > Underground piping for the high-pressure fire water system was installed and is maintained using the provisions of this standard for guidance only. The Licensee will satisfy the provisions of this standard for modifications to the facility except as documented and justified by the AHJ.

  For the reference to this standard see Section 7.1.1 of this license application.

- NFPA 30-2003, *Flammable and Combustible Liquids Code*

  The Licensee satisfies the requirements of this standard with the following exceptions/clarification:

  > Above ground storage tanks were installed using the provisions of this standard for guidance only. The Licensee will satisfy the provisions of this standard for modifications to the facility except as documented and justified by the AHJ.

  For references to this standard see Sections 7.1.1, and 7.3 of this license application.

- NFPA 51B-2003, *Standard for Fire Prevention During Welding, Cutting, and Other Hot Work*

  The Licensee uses the provisions of this standard as guidance for the review of hot work permitting.

  For the reference to this standard, see Section 7.1.6 of this license application.

- NFPA 55-2005, *Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks*

  The Licensee uses the provisions of this standard as guidance for the use of compressed gases.

  For the reference to this standard, see Section 7.3 of this license application.

- NFPA 72-2002, *National Fire Alarm Code*

  This NFPA standard was used as guidance for the installation of the system.

For the reference to this standard see Section 7.3.4 of this license application.

- NFPA 101-1991, *Life Safety Code*

  The Licensee uses the provisions of this standard as guidance for the review of emergency egress paths.

  For the reference to this standard see Section 7.2.3 of this license application.

- NFPA 232-1986 (and 232 AM), *Standard for the Protection of Records*

  The Licensee satisfies the provisions of this standard with the following exceptions/clarification:

  > As described in Section 11.7.1.8 of the licensing application, there are several acceptable methods for the storage of permanent records. If the NFPA 232 (or 232 AM) method of storage in 2-hour-rated containers is used, any exceptions to this standard will be documented and justified by the AHJ.

  For the reference to this standard see Section 11.7.1.8 of this license application.

- NFPA 801-1998, *Standard for Fire Protection for Facilities Handling Radioactive Materials*

  The Licensee will utilize this standard for any future modifications to the fire protection program as stated in Section 7.1.1 of the license application.

  For the reference to this standard see Section 7.1.1 of this license application.

1.4.6 Nuclear Regulatory Commission Guidance

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

  The Licensee 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*

  The Licensee utilized the provisions of this RG as guidance for PORTS Emergency Plan.

For references to this RG see Section 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. The Licensee 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*

  The Licensee 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 1, *Air Sampling in the Workplace*

  The Licensee 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*

  The Licensee 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.

- Regulatory Guide 1.109, Revision 1, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I*

  The Licensee satisfies the provisions of this RG to the extent applicable to Part 70 licensee.

  For references to this RG see Sections 9.2.2.1.2, and 9.2.2.2.2 of this license application.

- Regulatory Guide 1.159, Revision 0, *Assuring the Availability of Funds for Decommissioning Nuclear Reactors*

  This RG was used as a general reference and guidance document during development of this license application.

For the reference to this RG see Section 10.10.1 of this license application.

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

  This NUREG was used as a general reference and guidance document during the development of the Integrated Safety Analysis (ISA) and ISA Summary

  For references to this NUREG see Sections 3.1.2, 3.2, 3.3, 5.5, 6.14, 7.2.2, 7.6, 8.2, 9.2.3, and 9.4 of this license application.

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

  This NUREG was used as a general reference and guidance document during the development of the license application. This license application follows the format and guidelines of the NUREG.

  For references to this NUREG see Sections 1.0, 1.4, 3.2, 5.5, 6.14, 7.6, 8.2, 9.2.3, 9.4, 10.11, and 11.9 of this license application.

- NUREG-1601, *Chemical Process Safety at Fuel Cycle Facilities*

  This NUREG was used as a general reference and guidance document during the development of the license application.

  For the references to this NUREG see Section 6.14 of this license application.

- NUREG/BR-0006, *Instructions for Completing Nuclear Material Transaction Reports*

  This NUREG describes the requirements for reporting nuclear material transactions to the national database. 10 CFR 74.15 requires that instructions in this NUREG be followed.

  The Licensee satisfies the provision of this NUREG.

  For the reference to completion of Nuclear Material Transaction Reports, see Section 7.3 of the Fundamental Nuclear Material Control Plan (FNMCP) for the American Centrifuge Lead Cascade Facility.

- NUREG/BR-0007, *Instructions for the Preparation and Distribution of Material Status Reports*

  This NUREG describes the requirements for submitting material status reports to the national database. 10 CFR 74.13 requires that instructions in the NUREG be followed.

  The Licensee satisfies the provisions of the NUREG to the extent possible for uranium enrichment facilities.

  For the reference to this NUREG, see Sections 6.4 and 7.3 of the FNMCP for the Lead Cascade Facility.

- NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*

  Portions of this NUREG were used as a general reference and guidance document in the development of the accident analyses in the ISA.

  For the reference to this NUREG see Section 3.3 of this license application.

- NRC Information Notice No. 88-100: *Memorandum of Understanding between NRC and OSHA Relating to NRC-Licensed Facilities (53 FR 43950, October 31, 1988)*, December 23, 1988

  The Licensee has reviewed the information contained in this Information Notice.

  For the reference to this IN see Section 6.14 of this license application.

1.4.7 Other Codes, Standards, and Guidance

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

  The Licensee satisfies the provisions of this guidance document.

  For the reference to this guidance document see Section 4.7.4 of this license application.

- American Society for Nondestructive Testing Recommended Practice No. SNT-TC-1A, June 1980 Edition

  The Licensee satisfies the provisions of this recommended practice.

  For the reference to this recommended practice see Section 2 of the QAPD.

1.5 References

1. USEC 2002 Annual Report

2. U.S. Bureau of the Census, 2000, "Profiles of General Demographic Characteristics: 2000 Census of Population and Housing, Ohio", U.S. Department of Commerce, May 2001, Website: http://www.census.gov/prod/cen2000/dp1/2kh39.pdf

3. 329-10-002, ACP Memo dated October 15, 2010, "Worker and Transient Populations in and around PORTS DOE Reservation, as of October 2010", S. E. Keller

4. USEC-02, Application for United States Nuclear Regulatory Commission Certification, Portsmouth Gaseous Diffusion Plant, Safety Analysis Report

5. Ohio Department of Natural Resources, Website accessed October 21, 2002, http://www.dnr.state.oh.us/parks/parks/lkwhite.htm

6. U.S. Department of the Interior, U.S. Geological Survey, Reston, VA, Website: http://www.usgs.gov/index.html

7. Tetra Tech, Inc. correspondence, "Methodology for the 5-mile Population Grids," November 2002

8. CDR-2000-0001, Lead Cascade Conceptual Design Report, Revision 0, January 13, 2003

9. Lead Cascade Security Program, AET 02-0004, Steven A. Toelle letter to Mr. Martin J. Virgilio, dated July 3, 2002

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

2.0 ORGANIZATION AND ADMINISTRATION

The Licensee is committed to conducting operations at the Lead Cascade in a manner that protects the health and safety of workers and the public, protects the environment, and provides for the common defense and security. In order to meet these commitments, as well as others required for operation of the Lead Cascade, the Licensee maintains the following operations policy with respect to environmental, health, nuclear safety, safeguards, security, and quality to guide the day-to-day business activities of, and provide direction to, Lead Cascade personnel.

> The Licensee is responsible for safe operation of the Lead Cascade 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.

The Licensee has provided the management structure to ensure that this policy is effectively implemented and is responsible for the safe operation of the Lead Cascade. Programs and staff organizations are established for the environmental, health, safety, safeguards, security, and quality areas and are provided with sufficient resources to support safe operation of the Lead Cascade. Resources from the United States Enrichment Corporation (Corporation) at the Portsmouth Gaseous Diffusion Plant (PORTS) are utilized in a number of these programmatic areas to provide day-to-day functional support. Inter-company arrangements are in place with the Corporation to provide the necessary support. To the extent that the Licensee relies on existing programs and resources from PORTS, such programs and resources meet 10 *Code of Federal Regulations* Part 70 requirements.

The Licensee is responsible for the design, quality assurance (QA), refurbishment/construction, manufacturing, testing, start-up, operation and maintenance of the Lead Cascade. Preparation of some refurbishment/construction documents and portions of the refurbishment/construction itself are contracted to a qualified contractor. The Licensee staffs the Lead Cascade with qualified individuals to ensure a smooth transition from refurbishment activities to facility operations.

Managerial positions that have the principal responsibilities important to environmental, health, safety, safeguards, security, and quality for the Lead Cascade are described in this chapter. Their qualifications, responsibilities, and authorities are clearly defined in position descriptions that are accessible to affected personnel and the U.S. Nuclear Regulatory Commission (NRC) upon request.

Section 2.1 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 provide for the common defense and security from design through refurbishment/construction, start-up, and operation. Each manager has stop work authority for activities under their area of responsibility and if such authority is exercised, they must also concur with restart of those shutdown operations. If QA personnel exercise stop work authority, the Vice President, American Centrifuge must concur with restart.

Section 2.2 describes the management controls for maintaining the environmental, health, safety, safeguards, and quality programs and the administrative systems to control relationships and interfaces between programs.

Section 2.3 describes the Licensee's plans and the management controls for pre-operational testing and initial start-up of the Lead Cascade.

2.1 Organizational Commitments, Relationships, Responsibilities, and Authorities

The Lead Cascade management structure provides for line responsibility for safe operations with sufficient staff support to develop, communicate, and implement technical programs for various environmental, health, safety, safeguards, security, and quality areas.

Figure 2.1-1 shows the Lead Cascade organization for the safe operation of the Lead Cascade.

Day-to-day functional support for carrying out the requirements of the environmental, safety, health, safeguards, and security programs is provided by the Corporation, along with administrative services required to support overall facility operations. A description of the Corporation's gaseous diffusion plant (GDP) management structure and associated responsibilities is described in Reference 4. Services provided by the Corporation include those listed in Table 2.1-1. Table 2.1-1 identifies which Lead Cascade manager has decision making authority and responsibility for oversight of the major functional support areas provided by the Corporation. The Corporation also provides the necessary utilities (e.g., electricity, compressed air, cooling water, potable water, and sanitary sewage) to support operation of the Lead Cascade.

Minimum qualifications, functions and responsibilities for key staff positions are described below. The personnel responsible for managing the design, refurbishment/construction, manufacturing, and operation of the facility have substantive breadth and level of experience in their areas of expertise to successfully execute their responsibilities. These key positions are available as necessary to provide timely support in their respective functional area. Alternates are designated in writing and in accordance with procedural requirements to fulfill the responsibilities and authorities of these personnel during their absence.

Throughout this section, equivalent technical experience means the substitution of two years of nuclear industry experience for each year of college up to a total of three years. Additionally, 30-semester hours or 45-quarter hours from an accredited college or university may be substituted for the remaining one 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 are not 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 are evaluated on a case-by-case basis, documented, and approved by the appropriate Director or General Manager.

2.1.1 Vice President, American Centrifuge

The Vice President, American Centrifuge reports to the Senior Vice President and has overall responsibility for the American Centrifuge program's design, QA, refurbishment/construction, manufacturing, start-up, testing, operation, and maintenance of the Lead Cascade. The Vice President, American Centrifuge is responsible for the QA Program and for determining, the status, adequacy, and effectiveness of the Quality Assurance Program Description (QAPD).

The General Manager, American Centrifuge Plant Operations; Director, Regulatory and Quality Assurance; Director, Technology and Process Engineering; Director, Centrifuge Manufacturing; and Technical Director, American Centrifuge Project report to the Vice President, American Centrifuge and manage the activities in their area of responsibility.

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 management experience, which may be concurrent with the nuclear experience.

2.1.2 Director, Regulatory and Quality Assurance

The Director, Regulatory and Quality Assurance reports to the Vice President, American Centrifuge. This director has responsibility for the management of regulatory and QA functions. 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 Lead Cascade Facility. Although this individual works closely with the line management and key plant personnel, the Director, Regulatory and Quality Assurance is independent from production, plant operating cost, and production schedule concerns.

The Regulatory Manager and QA Manager reports to the Director, Regulatory and Quality Assurance and manage the activities in their area of responsibility.

The Director, Regulatory and Quality Assurance has, as a minimum, a bachelor's degree in engineering or the 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.

2.1.2.1 Regulatory Manager

The Regulatory Manager reports to the Director, Regulatory and Quality Assurance. The Regulatory Manager is responsible for regulatory oversight functions and commitment management. The Regulatory Manager, as delegated by the Director, Regulatory and Quality

Assurance and other line management, maintains the day-to-day interface with NRC representatives on matters of regulatory compliance. This manager 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 ensure that corrective actions and commitments are properly addressed and managed to facilitate compliance with the implementing policies and procedures. The Regulatory Manager is also responsible for the Nuclear Materials Control and Accountability (NMC&A) program.

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.

2.1.2.2 Quality Assurance Manager

The QA Manager reports to the Director, Regulatory and Quality Assurance. The QA Manager has the responsibility to exercise oversight of design, procurement, refurbishment/construction, manufacturing, testing, start-up, facility operations, and maintenance 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 QAPD, policies, and procedures. The QA Manager provides independent assessment and audit of Lead Cascade activities.

Although the QA Manager has direct access to the Vice President, American Centrifuge and interacts directly with the line management for QA matters, the QA Manager is independent from production, plant operating cost, and production schedule concerns. The QA Manager has access to information and participates (as desired) in any evaluations or discussions related to safety, safeguards, and quality.

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.

2.1.3 General Manager, American Centrifuge Plant Operations

The General Manager, American Centrifuge Plant Operations reports to the Vice President, American Centrifuge. The General Manager is responsible for the day-to-day safe operation of the facility, including operations and maintenance of the Lead Cascade facilities; overall responsibility for the Facility Safety Review Committee (FSRC), Nuclear Safety, and As Low As Reasonably Achievable program; compliance with applicable NRC regulatory requirements; and adherence to applicable policies and procedures. The General Manager also oversees activities of line management organizations that support the Lead Cascade refurbishment/construction activities, as applicable. The General Manager is the primary interface with NRC inspection personnel on matters of regulatory compliance within his/her scope of responsibility, and may delegate responsibility for this day-to-day interface to the Regulatory Manager.

The Technical Services Manager, Plant Services Manager, Business Services Manager, and Manager, Enrichment Operations report to the General Manager, American Centrifuge Plant Operations and manage the activities in their area of responsibility.

The General Manager, American Centrifuge Plant Operations 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.

2.1.3.1 Technical Services Manager

The Technical Services Manager reports to the General Manager, American Centrifuge Plant Operations. This manager is the design authority for the Lead Cascade and is responsible for engineering activities in support of facility operations. This includes configuration management; nuclear safety, including NCS and maintenance of the Integrated Safety Analysis (ISA); system engineering, and environmental, safety and health (ES&H), including radiation protection (RP).

The Technical Services 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.

2.1.3.1.1 Nuclear Safety Manager

The Nuclear Safety Manager reports to the Technical Services Manager. This manager is responsible for developing and implementing the nuclear safety program for the Lead Cascade, including technical oversight of nuclear safety, management of the ISA, safety analysis training, review of procedures involving fissile material operations, and assessments of nuclear safety program implementation. This manager is also responsible for direct management of the nuclear criticality safety (NCS) functions and administration of the NCS program on a day-to-day basis. These activities may include conducting assessments of program implementation; ensuring adherence to NCS evaluation requirements; review and approval of fissile material operations; review and approval of design changes that could affect or establish new fissile material operations; developing posting and labeling requirements; and NCS training requirements. This manager has direct access to the General Manager, American Centrifuge Plant Operations and the Vice President, American Centrifuge for nuclear safety matters.

The Nuclear Safety Manager has, as a minimum, a bachelor's degree in engineering, mathematics, or related science or equivalent technical experience, and four years nuclear experience, including six months at a uranium processing plant where NCS was practiced.

2.1.3.1.2 Engineering Manager

The Engineering Manager reports to the Technical Services Manager. This manager has responsibility for the engineering activities in support of plant operations, which includes maintaining the configuration management program and supporting procurement services.

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.

2.1.3.1.2.1 Configuration Management Program Manager

The Configuration Management Program Manager reports to the Engineering Manager. This manager has the responsibility for maintaining the configuration management program plan and overseeing the implementation of the program to ensure that the physical equipment and facilities; the drawings, specifications, and procedures; and the design/licensing basis for the facility are all maintained.

The Configuration Management Program 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.

2.1.3.1.3 System Engineering Manager

The System Engineering Manager reports to the Technical Services Manager. This manager has responsibility for the system engineering activities in support of plant operations, which includes providing engineering support and review of the design and modifications of items relied on for safety (IROFS).

The System 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.

2.1.3.1.4 Environmental, Safety, and Health Manager

The ES&H Manager reports to the Technical Services Manager. This manager is responsible for programmatic oversight of industrial hygiene, industrial safety, chemical process safety, Radiation Protection Program, waste management, and environmental monitoring.

The ES&H 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.

2.1.3.1.4.1 Radiation Protection Manager

The Radiation Protection Manager (RPM) reports to the ES&H Manager. The RPM is responsible for the RP Program and administration on a day-to-day basis, including providing guidance and direction for establishment and implementation of the RP Program and has the authority to deny access to radiological areas for personnel who do not adhere to radiological protection requirements. The RPM also has oversight of all radiological protection procedures in order to maintain the integrity of the RP Program. The RPM has direct access to the General Manager, American Centrifuge Plant Operations and the Vice President, American Centrifuge for radiation protection matters.

The RPM has, as a minimum, a bachelor's degree in engineering, health physics, RP, or the physical sciences or equivalent technical experience, and four years experience in RP, including six months at a uranium processing plant.

2.1.3.2 Plant Services Manager

The Plant Services Manager reports to the General Manager, American Centrifuge Plant Operations. This manager is responsible for the fire safety, emergency management, security, information technology, and facility services functions.

The Plant Services 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.

2.1.3.2.1 Fire Safety/Emergency Management Manager

The Fire Safety/Emergency Management Manager reports to the Plant Services Manager. This manager is responsible for the Fire Safety program; fire protection systems and services (i.e., including emergency and fire response, fire inspection, fire testing services, interpretation and application of applicable fire codes and standards); and emergency management.

The Fire Safety/Emergency Management 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.

2.1.3.2.2 Security Manager

The Security Manager reports to the Plant Services Manager. This manager is responsible for the strategic direction of the site security operations and programs for the Lead Cascade safeguards and security services. The Security Manager has direct access to the Vice President, American Centrifuge for security matters.

The Security Manager has, as a minimum, a bachelor's degree or equivalent technical experience, and four years security experience.

2.1.3.3 Business Services Manager

The Business Services Manager reports to the General Manager, American Centrifuge Plant Operations. This manager is responsible for finance, public affairs, human resources, procurement, training, procedures, and Records Management and Document Control (RMDC).

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

2.1.3.3.1 Training Manager

The Training Manager reports to the Business Services Manager. This manager is responsible for preparation, presentation, and documentation of employee orientations; and for technical and qualification training program development and implementation. This manager is also responsible for the development and implementation of the Procedures program and RMDC programs.

The Training 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.

2.1.3.3.2 Procurement Manager

The Procurement Manager reports directly to the Director, Procurement and Contracts, which is outside the American Centrifuge organization. However, this position is matrixed to the Business Services Manager and reports to this position for procurements related to the American Centrifuge Lead Cascade Facility. This manager is responsible for acquisition of equipment, materials, and services to support plant operations. The Procurement Manager ensures goods and services meet the specifications requested at a reasonable price and in a timely and efficient manner.

The Procurement manager has, as a minimum, a bachelor's degree in business or physical sciences or equivalent technical experience, and four years of nuclear experience.

2.1.3.4 Manager, Enrichment Operations

The Manager, Enrichment Operations reports to the General Manager, American Centrifuge Plant Operations. This manager is responsible for operations, maintenance, integrated planning, scheduling, and materials management.

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.

2.1.3.4.1 Operations Manager

The Operations Manager reports to the Manager, Enrichment Operations. This manager is responsible for the enrichment operations, production management, shift operations, and select repair of centrifuge machines.

The Operations 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, including six months at a uranium processing plant.

2.1.3.4.1.1 Process Area Managers

Process Area Managers report to the Operations Manager. The Process Area Managers are responsible for directing the activities of the Operations Shift Supervisors in the operation of the cascade, feed and withdrawal, gas test, and plant utilities processes and facilities. This includes activities such as ensuring the safe operation of the uranium hexafluoride (UF_6) processes; proper receipt, storage, handling, and on-site transportation of UF_6; machine installation and pump down; and integrated system testing.

The Process Area Managers have, as a minimum, a high school diploma or satisfactory completion of the General Educational development test, and three years of industrial/chemical/nuclear plant operations, maintenance, engineering, or support experience.

2.1.3.4.1.2 Operations Shift Supervisors

Operations Supervisors report to the Process Area Manager. As the senior manager on shift (one per shift), the Operations Shift Supervisor represents the General Manager, American Centrifuge Plant Operations and has the authority and responsibility to make decisions, as necessary, to ensure safe operations. The Operations Shift Supervisors are responsible for accumulation and dissemination of information regarding plant activities to the Incident Commander during emergencies. The Operations Shift Supervisors are also responsible for directing the operation of systems within the facilities necessary to support enrichment operation. The Operations Shift Supervisors authorize the restart of equipment that has been shutdown in a routine fashion when the prerequisites and limitations of the associated operating procedure are met. The Operations Shift Supervisor is responsible for providing operational support of centrifuge machine assembly, transport, installation, pump down, integrated system testing, start-up, operation, and select repair. The Operations Shift Supervisors also direct the operation of systems within the Lead Cascade facilities, necessary to support Lead Cascade operation.

Operations Shift Supervisors have, as a minimum, a high school diploma or satisfactory completion of the General Educational Development test, and three years of industrial/chemical/nuclear plant operations, maintenance, or engineering experience. Operations Shift Supervisors must have one year of supervisory experience or completion of a supervisory training course.

2.1.3.4.2 Maintenance Manager

The Maintenance Manager reports to the Manager, Enrichment Operations. This manager is responsible for overall plant maintenance activities at the Lead Cascade to ensure safe and reliable performance of equipment and facilities using preventive, predictive, and corrective maintenance techniques, with the exception of centrifuge machines. These services include equipment and system troubleshooting; maintenance of logs and records; work control and planning to initiate, screen, evaluate, and prioritize maintenance work; and coordinating shop maintenance activities. This manager is also responsible for integrated planning, scheduling, and materials management. This includes managing daily work control activities; maintenance of an integrated work schedule; and coordinating development of work control guidelines.

The Maintenance 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, including six months at a uranium processing plant.

2.1.3.4.2.1 Maintenance Work Center Supervisors

Maintenance Work Center Supervisors report to the Maintenance Manager. The Maintenance Work Center Supervisors are responsible for directing the activities of the Maintenance Shift Supervisors in the performance of preventive, predictive, and corrective maintenance and to provide support services on facilities and equipment, with the exception of centrifuge machines. These activities may include maintenance of electrical equipment; electronic and pneumatic instrumentation and controls; computers and programmable controllers; and mechanical maintenance, such as valve, pump, and mechanical equipment repair and replacement.

Maintenance Work Center Supervisors have, as a minimum, a high school diploma or satisfactory completion of the General Educational Development test, and three years of industrial/chemical/nuclear plant operations, maintenance, or engineering experience. Maintenance Work Center Supervisors must have one-year supervisory experience or completion of a supervisory training course.

2.1.4 Plant Shift Superintendent (PORTS)

The Plant Shift Superintendent (PSS) reports to the PORTS Shift Operations Manager.

The PSS is responsible for accumulation and dissemination of information regarding site activities, serving as or designating an incident commander during emergencies, and making notification of events. The PSS has the authority and responsibility to make decisions as necessary to ensure safe site operations, including stopping work. The PSS provides the Lead Cascade with a centralized point for incident identification, screening, and reporting. The PSS's responsibilities for the Lead Cascade are consistent with those exercised at PORTS for emergency response.

The PSS has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience and four years experience at a GDP, or a high school diploma plus 12 years experience at a GDP.

2.1.5 Director, Technology and Process Engineering

The Director, Technology and Process Engineering reports to the Vice President, American Centrifuge. This director is responsible for providing technical expertise and is the subject matter expert for the engineering of the core centrifuge technologies involving the centrifuge machine and Feed and Withdrawal Systems.

The Centrifuge Machine Engineering Manager reports to the Director, Technology and Process Engineering and manage the activities in his/her area of responsibility.

The Director, Technology and Process Engineering 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.

2.1.5.1 Centrifuge Machine Engineering Manager

The Centrifuge Machine Engineering Manager reports to the Director, Technology and Process Engineering. This manager is responsible for providing technical expertise for the centrifuge machine for the Lead Cascade and is responsible for machine dynamics and balancing; motor/machine drive unit design and machine instrumentation; materials engineering; reliability engineering; and Data Acquisition System support.

The Centrifuge Machine 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.

2.1.6 Director, Centrifuge Manufacturing

The Director, Centrifuge Manufacturing reports to the Vice President, American Centrifuge. The Director, Centrifuge Manufacturing is responsible for providing technical administration and direction to strategic suppliers for manufacturing and delivery activities of centrifuge components and systems and for the assembly of centrifuge machines.

The Director, Manufacturing 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.

2.1.7 Technical Director, American Centrifuge Project

The Technical Director, American Centrifuge Project reports to the Vice President, American Centrifuge and has responsibility for overall Technical Authority for the ACP. The Technical Director is responsible for integration of engineering activities across all ACP functional organizations to ensure engineering activities are coordinated and aligned with overall project priorities. The Technical Director is also responsible for the management of the Risk and Reliability program.

Individuals responsible for technical work will be managed by their respective functional manager, but will have matrix reporting responsibility to the Technical Director for matters under his/her cognizance.

The Technical Director has, as a minimum, a bachelor's degree in engineering or equivalent technical experience and 15 years experience managing complex major engineering project activities.

2.1.8 Shift Crew Composition

The minimum operating shift crew consists of an Operations Shift Supervisor, one Operations Technician, and one Radiation Protection/Industrial Hygiene technician.

Table 2.1-1 Responsibilities for Functional Support Provided by Portsmouth Gaseous Diffusion Plant

Lead Cascade manager responsible for oversight of functional support

	Regulatory Manager	Plant Services Manager
Functional Support Area provided by the Corporation	Nuclear Materials Control and Accountability	Fire Safety Emergency Management

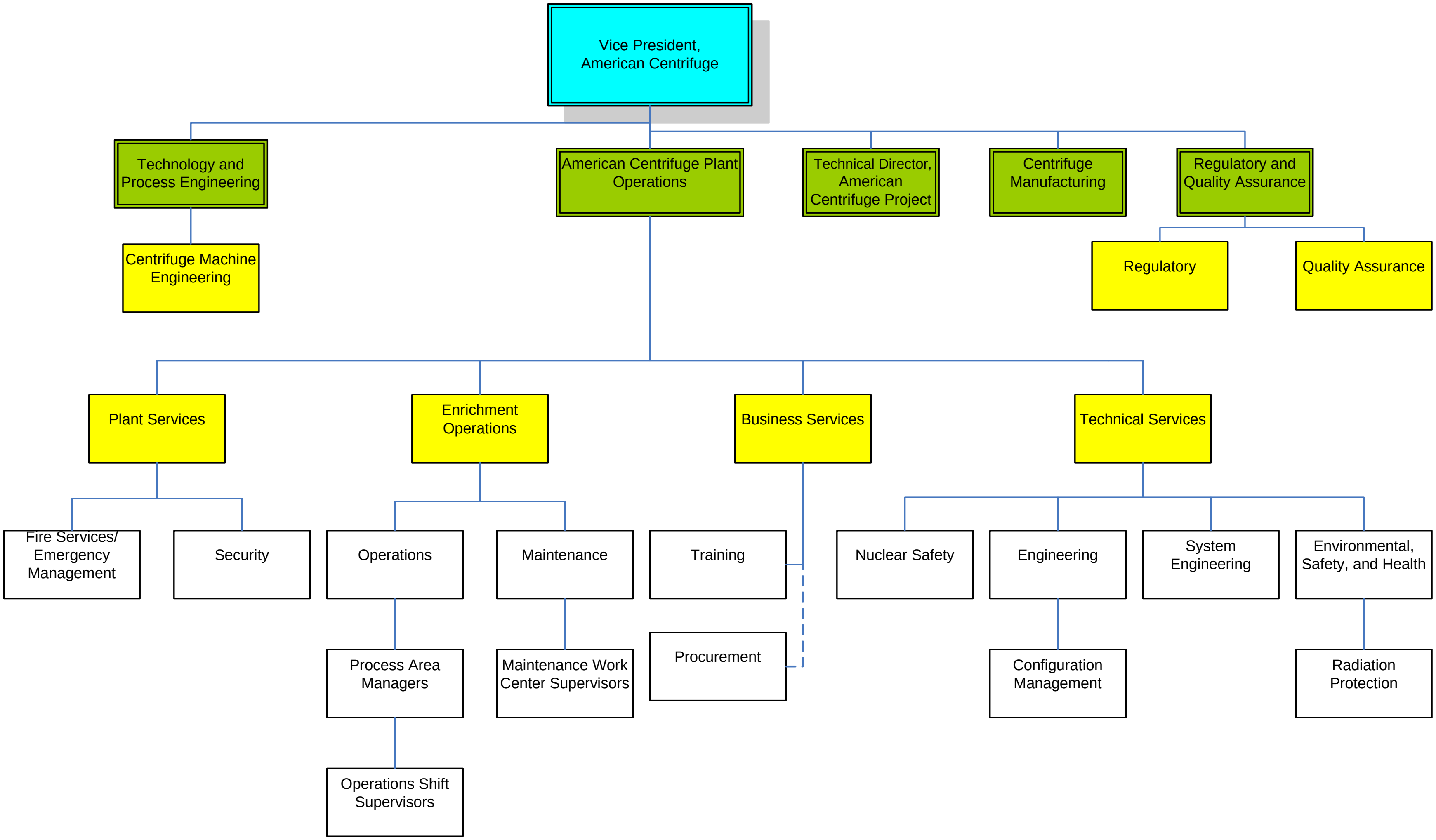

Figure 2.1-1
Lead Cascade Organization

2.2 Management Controls

The Licensee has established management systems with associated policies, administrative procedures, and management controls to ensure: the Lead Cascade 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 the common defense and security. Management controls have been established to maintain configuration management of the Lead Cascade. These controls are described in Section 11.1 of this license application. Organizations with environmental, health, safety, safeguards, security, and quality responsibilities have been established with a reporting chain, independent from the operations organization. Effective lines of communication and authority among the organizations involved in the engineering, environmental, safety, and health, and operations functions of the facility are clearly defined.

The management controls established by the Licensee for the Lead Cascade include policies, management systems, and administrative procedures that are communicated to Lead Cascade personnel. Policies related to the protection of health and safety of workers and the public, protection of the environment, and providing for the common defense are discussed in pertinent sections of this license application. Activities that are essential for effective implementation of the environmental, safety, and health functions are documented in approved, written procedures, prepared in compliance with a document control program. Procedure development and document control are described in Section 11.4 of this license application and Sections 5 and 6 of the QAPD.

Management measures required to ensure the availability and reliability of IROFS are described in Chapter 11.0 of this license application. Controls specific to Lead Cascade programs are identified in the QAPD, Fundamental Nuclear Material Control Plan, and Security Program for the Lead Cascade Facility.

The commitment tracking and corrective action management systems are integrated to prioritize Lead Cascade actions consistent with their safety and safeguards significance. Any person working in the facility may report potentially unsafe conditions or activities by submitting a problem report. Reported concerns are investigated, assessed, and resolved as described in Section 11.6 of this license application.

Where safety, security, or safeguards might be adversely impacted by cost or schedule considerations, it is the policy of the Licensee to subordinate cost and schedule considerations to ensure adequate treatment of safety and safeguards in full compliance with applicable regulatory requirements.

The integration of Lead Cascade operations and the various programs and requirements is 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 FSRC
- 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 is provided by the QA Organization.

Written interface agreements exist with offsite emergency resources such as fire, police, ambulance/rescue units, and medical services. These interface agreements are addressed in more detail in the PORTS Emergency Plan.

2.2.1 Facility Safety Review Committee

The FSRC performs multi-discipline reviews of day-to-day and proposed Lead Cascade activities to ensure that these activities are and/or will be conducted in a safe manner. The FSRC advises the General Manager, American Centrifuge Plant Operations on matters related to RP, Nuclear Safety, Chemical Safety, Fire Safety, and Environmental Protection. The specific membership, qualifications, meeting frequency, quorum, functions, responsibilities, and required records are provided in a facility procedure. Auditing and oversight of FSRC activities is the responsibility of the QA Manager.

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

2.3 Pre-operational Testing and Initial Start-up

Specific plans have been established to ensure the safe and efficient turnover, testing, and start-up of Lead Cascade centrifuge machines, equipment, and support systems. These plans cover the transition from the refurbishment/construction phase to the operations phase of the Lead Cascade Project.

The Technical Services Manager is responsible for development and implementation of testing to provide for the turnover and acceptance of equipment and systems from contractors/vendors to the Licensee.

The Operations Manager is responsible for the development and execution of the Integrated Systems Test Plans (ISTPs). The Technical Services Manager may assist in the

development of ISTPs. The ISTPs demonstrate the proper operation of completed systems to ensure the systems meet their intended design functions. This manager is also responsible for the testing, initial start-up, and operation of the centrifuge machines, equipment, and support systems. Documentation of testing is maintained in accordance with RMDC requirements, and is available for NRC review.

2.3.1 Pre-operational Testing Objectives

The overall objectives of the pre-operational test program are to ensure that the Lead Cascade facilities and systems, including the IROFS:

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

2.3.2 Turnover, Functional, and Initial Start-up Test Program

The refurbishment contractor is responsible for completion of as-built drawing verification, purging/flushing, cleaning, hydrostatic or pneumatic testing, system turnover, and initial calibration of instrumentation in accordance with design and installation specifications. As systems or portions of systems are turned over to Operations, acceptance testing is performed in accordance with established schedules. The Technical Services Manager is responsible for coordination of turnover and acceptance testing for the Lead Cascade.

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 contractor prior to turnover the Licensee. The testing that is performed is commensurate with the system or component's quality level and is principally associated with IROFS, but may also include other tests on systems or components that the Licensee deems appropriate for financial, reliability, or other reasons. Integrated systems tests include the testing that is necessary to demonstrate that the facility, system, or component is capable of performing its intended function. The Operations Manager is responsible for the execution of the ISTPs for the Lead Cascade. The integrated systems tests are performed following completion of construction; flushing; hydrostatic or pneumatic testing; system turnover; and initial calibration of required instrumentation. Scheduling of the testing is such that it generally occurs prior to UF_6 introduction. Other pre-operational tests, not required prior to UF_6 introduction, may be performed following introduction of UF_6 to the process system.

2.4 References

1. NR-2605-0001, Gas Centrifuge Quality Assurance Program Description

2. NUREG-1324, *Proposed Method for Regulating Major Materials Licensees*

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

4. USEC-02, Application for United States Nuclear Regulatory Commission Certification, Portsmouth Gaseous Diffusion Plant, Safety Analysis Report

5. USEC-02, Application for United States Nuclear Regulatory Commission Certification, Portsmouth Gaseous Diffusion Plant, Emergency Plant

3.0 INTEGRATED SAFETY ANALYSIS AND INTEGRATED SAFETY ANALYSIS SUMMARY

The requirements in 10 *Code of Federal Regulations* (CFR) 70.62(c) specify that an Integrated Safety Analysis (ISA) of the appropriate level of detail for the complexity of the process involved will be conducted and maintained. An ISA Summary is required by 10 CFR 70.65(b). Accordingly, the Licensee has conducted an ISA and prepared an ISA Summary for the American Centrifuge Lead Cascade Facility (U). The ISA identifies potential accident sequences in the facility's operations, designates items relied on for safety (IROFS) to either prevent such accidents or mitigate their consequences to an acceptable level, and describes management measures to provide reasonable assurance of the availability and reliability of IROFS.

The ISA Summary is a synopsis of the results of the ISA and contains the information required by 10 CFR 70.65(b). Neither the ISA nor the ISA Summary is incorporated as part of the license. The ISA and ISA Summary documents describe the methods and criteria utilized in the safety analysis and describe the qualification of the team performing the ISA.

3.1 Safety Program and Integrated Safety Analysis Commitments

3.1.1 Process Safety Information

The Chemical Process Safety program is described in Chapter 6.0 of this license application. Consistent with this program, the Licensee compiles and maintains an up-to-date database of process-safety information. Written process-safety information is used in updating the ISA and in identifying and understanding the hazards associated with the processes. The compilation of written process-safety information includes information pertaining to:

- The hazards of materials used or produced in the process, which includes information on chemical and physical properties (e.g., toxicity, acute exposure limits, reactivity, and chemical and thermal stability) such as those included on Material Safety Data Sheets (meeting the requirements of 29 CFR 1910.1200(g));

- Technology of the process, which includes a block flow diagram or simplified process flow diagram, a brief outline of the 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 (e.g., interlocks, detection, or suppression systems), electrical classification, and relief system design and design basis; and

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

3.1.2 Integrated Safety Analysis

An ISA for operation of the Lead Cascade was conducted in accordance with the guidance provided in NUREG-1513, *Integrated Safety Analysis Guidance Document* and the requirements of 10 CFR 70.62(c). The ISA is a collection of document available on site for U.S. Nuclear Regulatory Commission (NRC) review.

The ISA contains the appropriate details for each process, such that it identifies:

- Radiological hazards;
- Chemical hazards that could increase radiological risk;
- Facility hazards that could increase radiological risk;
- Chemical hazards from materials involved in processing licensed materials;
- 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.

Should new processes be necessary in the Lead Cascade, an evaluation of appropriate detail would be performed and the ISA and ISA Summary would be updated as required.

The Licensee maintains the ISA, its supporting documentation, and the ISA Summary so that it is accurate and up-to-date by means of a suitable configuration management system, described in Section 11.1 of this license application. Lead Cascade procedures specify the criteria for changing the ISA. Changes to the facility are evaluated using a change process that meets the requirements of 10 CFR 70.72. Changes to the ISA Summary are submitted to the NRC in accordance with 10 CFR 70.72(d)(1) and (3). The ISA accounts for any appropriate changes made to the facility or its processes (e.g., changes to the site, operating procedures, or control systems). Any facility change, operational change, or change in the process safety information that may alter the parameters of an accident sequence are evaluated by means of the ISA methods. The Licensee evaluates proposed changes to the facility or its operations by means of the ISA methods and designates new or additional IROFS, along with appropriate management measures, as required. The Licensee also evaluates the adequacy of existing IROFS and associated management measures and makes any required changes prior to making changes to the facility and/or its processes. If a proposed change results in a new type of accident sequence (e.g., different initiating event or significant changes in the consequences) or increases the consequences and/or likelihood of a previously analyzed accident sequence within the context of 10 CFR 70.61,

the Licensee evaluates whether changes to existing or additional IROFS, or associated management measures are required. For any changes that require prior NRC approval under 10 CFR 70.72, the Licensee would submit an amendment request in accordance with 10 CFR 70.34 and 70.65.

The Nuclear Safety Manager is responsible for maintaining the ISA and ISA Summary (i.e., reviewing proposed changes, performing analyses, and ensuring implementation of required updates). The Regulatory Manager is responsible for submitting the required changes to the NRC.

Suitably qualified personnel update and maintain the ISA and ISA Summary. The ISA Team for a process consists of individuals who are knowledgeable in the Lead Cascade's ISA methods and the operation, hazards, and safety design criteria of the particular process being evaluated. Personnel with appropriate experience and expertise in engineering and process operations are utilized in the maintenance and updating of the ISA. These individuals are trained in the facility's ISA methods and procedures are used to implement the ISA process. For any revisions to the ISA, personnel having qualifications similar to those of ISA Team members who conducted the original ISA are used.

3.1.3 Management Measures

Lead Cascade IROFS are identified in the ISA and ISA Summary. Management measures are utilized to maintain the IROFS so that they are available and reliable. Management measures are the principal mechanism by which the reliability and availability of each IROFS is ensured. They 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 Lead Cascade 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 summary is submitted for review (separate from this license application) as a document titled, Integrated Safety Analysis Summary for the American Centrifuge Lead Cascade Facility (U).

3.3 References

1. Integrated Safety Analysis Summary for the American Centrifuge Lead Cascade Facility (U)

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

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

4. NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*

4.0 RADIATION PROTECTION

This chapter describes the Lead Cascade Radiation Protection (RP) Program for keeping occupational radiation exposures and radioactive contamination 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 RP Program is based on the well-established Portsmouth Gaseous Diffusion Plant (PORTS) program and is implemented through written procedures. The Technical Services Manager is responsible for the Lead Cascade RP Program. Responsibilities of the Radiation Protection Manager (RPM) described in this chapter are carried out by the RPM or designee. The Licensee's program for minimizing and monitoring doses to the public and the environmental are discussed in Chapter 9.0 of this license application.

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 by the RPM and submitted to the ES&H Manager for further processing, as part of the annual review process. Any resulting changes to the Radiation Worker Training module are also implemented.

4.2 ALARA Program

In accordance with 10 CFR 20.1101, the Licensee has established an RP Program for the Lead Cascade designed to protect personnel entering the Lead Cascade facilities 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 this policy rests with the Vice President, American Centrifuge. The General Manager, American Centrifuge Plant Operations has the overall responsibility and authority for the ALARA Program. The RPM is responsible for establishing and implementing the ALARA Program.

4.2.1 ALARA Committee

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

The ALARA Committee:

- Communicates management's commitment to the ALARA Program;
- Monitors the implementation of the ALARA Program and serves as advisor to Lead Cascade management for maintaining occupational dose and environmental dose in accordance with ALARA principles; and
- Reviews, for the purpose of occupational dose and environmental dose reduction, proposed designs, practices, selected suggestions, and selected project schedules.

The ALARA Committee also:

- Establishes the annual exposure goals;
- Provides recommendations to line management and/or the Facility Safety Review Committee when determined appropriate, regarding procedural, equipment, or design changes that could have a significant impact on personnel radiation exposure; and
- Forms subcommittees or assigns individuals to undertake special studies or conduct ALARA reviews that will be documented and presented to the ALARA Committee with any recommendations.

Membership consists of persons from various functional disciplines of the Lead Cascade who have the necessary competence and experience to perform the functions of the committee. Standing committee members are the RPM who serves as the chairperson, the vice-chairperson who is appointed by the RPM, the Technical Services Manager, Operations Manager, Maintenance Manager, Regulatory Manager, and an operations technician and/or maintenance technician. Participation from other functional disciplines may vary depending on the issue of concern. The committee chairperson, or designee, is responsible for requesting appropriate functional representation. Committee members may designate an alternate to attend committee meetings in their place.

The ALARA Committee meets at least annually and as directed by the chairperson. A quorum consists of five standing committee members or their alternates. Ad hoc subcommittees may be established for special studies or reviews pertinent to committee-related issues.

The chairperson ensures those functions of the committee and tasks are properly executed. Minutes are provided to the General Manager, American Centrifuge Plant Operations. The Committee issues special reports prepared upon request of Lead Cascade management, or as determined by the chairperson.

The committee reviews matters that have or may have an impact on contamination control and/or ALARA. The ALARA Committee reviews the ALARA program and the review includes an evaluation of the results of audits made by the RP organization, reports of radiation levels in the facility, contamination levels, employee exposures, and effluent releases, etc. The review 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 Lead Cascade 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 General Manager, American Centrifuge Plant Operations and Vice President, American Centrifuge for radiological concerns. The RPM and designee are required to have the technical competence and experience to establish RP programs and the management capability to direct the implementation and maintenance of RP programs.

The PORTS Health Physics (HP) Group provides radiological protection support to the facility, is independent of the organizations responsible for production, and has an equivalent reporting level. 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 Lead Cascade 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 suspect that the initiation or continued performance of a job, evolution, or test will result in the violation of approved RP requirements.

4.4 Written Procedures

4.4.1 Procedures

The RP Program is implemented using procedures. The procedures are prepared consistent with the requirements of 10 CFR Part 20 and are approved, maintained, and adhered to for operations involving personnel radiation exposure and toxicological exposure to soluble uranium. The procedures are reviewed and revised as necessary to incorporate any facility or operational changes, including these initiated by changes to the Integrated Safety Analysis. These procedures are prepared, maintained and made available to appropriate personnel at the facility as described in Section 11.4 of this license application.

4.4.2 Radiation Work Permits

Radiation Work Permits (RWPs) are a basic implementing tool by which radiological controls are established. RWPs provide information to the worker concerning protective clothing, job/task identification, and special instructions such as radiological hold points. Radiological surveys that supplement RWPs provide information regarding radiation and contamination levels.

RWPs are required for work activities in Contamination Areas (CAs), High Contamination Areas (HCAs), Airborne Radioactivity Areas (ARAs), Radiation Areas (RAs), High Radiation Areas (HRAs) and other areas as required by HP. Qualified HP personnel are authorized to approve, issue, update, revise, and close RWPs. The RPM may exempt the requirement for an RWP in certain RAs as specified in approved procedures.

The limits established for contamination control (surface and airborne) are based on the toxicity of soluble uranium. The contamination control program, of which RWPs are a part, is designed to ensure that the inhalation or ingestion of soluble uranium is below the limits stated in 10 CFR 20.1201(e).

An RWP may be issued for any period up to one year, based on the stability and predictability of changes in the radiological conditions of the work area. RWPs are normally closed upon job completion. HP may close an RWP at any time.

Radiological surveys are reviewed to evaluate the adequacy of RWP requirements. RWPs are updated or closed and reissued if radiological conditions change to the extent those protective requirements need to be modified.

HP management reviews the RWP closure package to ensure appropriate actions have been taken.

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 Lead Cascade 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 described in Section 11.3.5 of this license application. The Radiological Worker Training Program outlines the requirements of 10 CFR 19.11 and 19.12 and the workers responsibilities under the Radiation Protection Program. The Radiation Worker Training program is described in Section 11.3.6 of this license application.

4.5.1 Visitor Site Access Orientation

Visitors review basic information related to the site and hazards present at the Lead Cascade. Visitors granted access to the Lead Cascade restricted areas are escorted by trained radiological workers.

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 Lead Cascade restricted area, radiological worker qualification is required. Radiation Worker Training is a biennial training requirement.

4.5.4 Health Physics Technician

HP Technicians and their managers are qualified and trained in accordance with an approved qualification standard and is delivered consistent with the 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. 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, facility-specific information, and on-the-job training. Passing a final comprehensive written examination is required. The training program ensures personnel are proficient in radiation measurements, characterization of radiological conditions, release monitoring, and personnel monitoring. Formal remediation protocols are utilized.

Entry-level prerequisites are established to ensure that HP Technicians meet minimum standards for education. Task qualification for entry-level positions may be used until formal training is completed.

Following initial qualification, HP Technicians are requalified every two years. The requalification process requires successful completion of a comprehensive written examination. The written examination may be waived for personnel with National Registry of Radiation Protection Technologist certification. Personnel, who maintain qualifications as HP Technicians, satisfy the requirements of Radiation Worker Training.

HP Technician managers maintain qualifications as HP Technicians.

4.6 Ventilation and Respiratory Protection Programs

Lead Cascade ventilation systems are described in Section 1.1.2.7.5 of this license application.

4.6.1 Ventilation

In addition to general ventilation systems, portable ventilation units may be employed for short duration jobs when the unprotected worker could potentially exceed 0.8 Derived Air Concentration (DAC)-hours of exposure. These local ventilation units are equipped with high efficiency particulate air (HEPA) filters and designed to re-circulate and discharge room air at low velocities. Activities where these units may be employed are also subject to approval by Nuclear Criticality Safety.

When used for radiological protection purposes, the portable HEPA filtered ventilation units differential pressure is checked per operating procedure. The operating differential pressure range is based on manufacturer's recommendations or as specified in the technical design basis. HEPA filter systems, both fixed and portable, are efficiency tested in accordance with American Society of Mechanical Engineers (ASME) N510-1989, *Testing of Nuclear Air-Treatment Systems,* as it applies to radiological contaminants likely to be found at the Lead Cascade. Portable HEPA filter units use is normally specified on the RWP.

HEPA filter systems are utilized to perform specific functions. HEPA filter systems required to implement ALARA principles and to control workers exposure 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 "glove boxes" are used for Lead Cascade activities, 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. There is a written Lead Cascade policy statement on respirator usage following 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, respiratory protection may not be used due to physical limitations, such as heat stress, or the potential for significantly increased external exposure with approval of the RPM. 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.)

4.7 Radiation Surveys and Monitoring Program

The Radiation Surveys and Monitoring Programs are based on the requirements of 10 CFR Part 20 and ALARA principles. Written procedures are prepared for the elements of the Radiation Survey and Monitoring Programs discussed in this section. Deficiencies associated with surveys and monitoring program or results that exceed the administrative control levels are dispositioned in accordance with the Quality Assurance Program Description and the Corrective Action Program, described in Section 11.6 of this license application.

4.7.1 Surveys

The radiological survey program consists of routine, work support, and material release (refer to Section 4.8.2.4 below) surveys. Surveys are conducted to support facility activities in a manner that ensures radiological hazards associated with each activity are properly identified, and relative radiation levels and concentrations of radioactive material are determined. Radiological surveys for the purposes of establishing personnel protection equipment or for posting requirements are performed by qualified HP personnel. Decontamination is performed as appropriate considering the gained benefit from waste minimization, ALARA principles and worker access.

The routine survey program involves surveys of the facility to determine workplace radiological conditions, effectiveness of contamination control measures, and proper identification and posting of radiological hazards. Routine survey frequencies are established based on the stability of operations as demonstrated by the consistency of survey results. Areas within the facility are categorized and scheduled for survey commensurate with their relative radiological hazard and contamination potential. Survey frequencies are based on area occupancy, potential for spread of contamination, and process knowledge. The routine survey program is reviewed annually by the RPM, documented, maintained, and modified to reflect changes in radiological conditions. Table 4.7-1 provides the contamination survey program frequencies for Lead Cascade areas.

In the event that large areas of removable contamination are identified on accessible surfaces exceeding the levels specified in Table 4.6-1, the area will be re-posted as a CA or HCA and actions taken to locate the source of contamination. If access is required to the area, decontamination of the area is initiated as soon as practical with consideration of ALARA principles.

Work support surveys are a fundamental element of the RWP process. In-process surveys are conducted as necessary to verify radiological conditions at various points in the work activity and to ensure exposure potentials are maintained in accordance with the ALARA principle. When required by work activities, surveys are conducted by qualified personnel to support decontamination efforts and the release of tools, equipment, and waste material from the work area.

4.7.2 Personnel Monitoring

Both U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Energy (DOE) regulated sources of radiation and radioactive materials are interspersed on the PORTS

reservation. There is also a frequent moving of personnel from a Licensee contractor or sub-contractor staff to a DOE contractor or sub-contractor staff. This situation makes separation of personnel exposure between NRC and DOE regulated sources impractical.

To comply with the personnel monitoring requirements of 10 CFR 20.1502 and the reporting requirements of 10 CFR 19.13, 20.2106 and 20.2206, the Licensee tracks exposures for personnel issued National Voluntary Laboratory Accreditation Program (NVLAP)-accredited dosimeters regardless of whether the exposure is from an NRC or DOE regulated source. Whenever worker notification is required by 10 CFR 19.13, the individual's "total exposure" while on the PORTS site is reported without differentiating between exposure from NRC-regulated sources and DOE regulated sources.

The established personnel monitoring program consists of the following:

- An Administrative Control Level (ACL) of 500 millirem (mrem) per year Total Effective Dose Equivalent per person;
- The intake limit for soluble uranium is set at 10 mg per week;
- Personnel dosimeters to measure the external exposure of personnel;
- Analysis of personnel occupational exposure and maintenance of exposure records; and
- A network of Fixed Nuclear Accident Dosimeters (FNADs) is situated in the Lead Cascade area. Dosimeters in the FNADs are processed by a NVLAP accredited dosimeter reader. The FNADs also serve as area monitors.

Personal dosimeters are also evaluated for neutron dose. In addition, permanent site personnel are provided an indium foil that can be evaluated for neutron activation. If the indium foil indicates exposure to a neutron flux exceeding 10 rads, the dosimeter is read and/or biological materials of personnel may be evaluated.

4.7.3 External

Persons requiring radiation exposure monitoring per 10 CFR 20.1502(a) wear beta-gamma-sensitive dosimeters which are processed and evaluated by a processor holding current NVLAP accreditation from the National Institute of Standards and Technology (NIST). Dosimeters are exchanged at least quarterly (+/-2 weeks) unless authorized in writing by the RPM. The dosimeters may be supplemented, as appropriate, by other types of dosimeters (e.g., finger rings, direct-reading dosimeters, and neutron dosimeters) and by 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 General

Manager, American Centrifuge Plant Operations. 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, 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 General Manager, American Centrifuge Plant Operations. Investigations to determine cause, assess the exposure, and document the results are specified by procedure.

External dosimetry results are reviewed by HP to determine any unusual trends or exposures. 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. Dose reports are completed as required for personnel monitored in accordance with 10 CFR 20.1502(a). 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 employees, subcontractors, and visitors must not exceed the 10 CFR Part 20, Subpart C limits. The Licensee requires current year exposure history of an occupational worker as required by 10 CFR 20.2104.

Personnel declaring pregnancy are advised to keep radiation exposure to an embryo or fetus in accordance with the ALARA principle during the entire gestation period. The Licensee 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 Lead Cascade are such that renal toxicity limitations are the limiting conditions for health effects. Historical experience at the gaseous diffusion plants (GDPs) indicates that exposure derived from low level chronic exposure is less than two percent of the annual radiation exposure limits specified in 10 CFR 20.1201.

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 gives a 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 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 Licensee 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.

Urinalysis results are reviewed by HP to determine unusual trends. 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 Lead Cascade 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, air samples are routinely 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.

Air sample flow measurement devices are calibrated under standard laboratory conditions at least annually. The NIST traceable standards used have accuracy and precision of 20 percent or better. Lapel samplers are calibrated as described by use procedure.

Table 4.7-1
Routine Contamination Survey Frequencies

Area Surveyed	Survey Frequency
Lead Cascade Uranium Processing Area	Yearly[a]
Contaminated Maintenance Areas	Quarterly
Contamination Control Zones (CCZ)	Quarterly
Lead Cascade Lunchrooms/Breakrooms	Note c
Permanent Boundary Control Stations (BCS)[b]	Weekly
Lead Cascade Change Rooms	Monthly
Lead Cascade UF_6 Sample Handling Laboratories	Monthly[a]

a. Localized area surveys are taken following an indication of release and during maintenance activities.

b. When personnel contamination is detected at the BCS, the ensuing follow-up activities include a physical survey of the BCS.

c. Surveys are performed daily during normal facility working days (i.e., Monday through Friday). Weekends and facility holidays are excluded.

Table 4.7-2
Bioassay Program

Urine Bioassay Capabilities	Comment
Workers Participation	Selected based on work locations
Frequency of Urine Monitoring	Monthly[a]
Routine Urine Sample Volume	Single void sample, between 60 and 100 mL
Primary Uranium Analysis Methods	Fluorimetry or Inductively Coupled Plasma (ICP) Mass Spectroscopy
ICP Mass Spectroscopy Minimum Detectable Concentration	<0.006 μg/L ^{235}U <0.015 μg/L ^{238}U
Fluorimetry Minimum Detectable Concentration	5 μg/L Total Uranium

Additional Analytical Capabilities	
Alpha Spectroscopy	0.1 pCi/sample[b]
Uranium Alpha with Proportional Counter	40 dpm/L Total Uranium in urine
Invivo Lung Counting	0.2 nCi ^{235}U 4 nCi ^{238}U
Dose Assessment Software	INDOS (Routine Analysis) CINDY (Developmental and Special)

a. Samples scheduled for submission every four weeks.

b. Equipment also used for loose contamination and airborne radioactivity samples for characterization efforts.

Table 4.7-3
Internal Dosimetry Program Action Levels

Bioassay Technique	Frequency	Action Level	Actions to be Taken
Urinalysis Routine	Monthly[a]	5 µg U/L	Resample to confirm result and determine intake[b]
	Monthly	20 µg U/L	Restrict individual and resample to determine intake[b]
Urinalysis Special	2-6 hours after intake	5 µg U/L	Resample to confirm result and determine intake[b]
		300 µg U/L	Restrict individual and resample to determine intake[b]
	16-30 hours after intake	5 µg U/L	Resample to confirm result and determine intake[b]
		50 µg U/L	Restrict individual and resample to determine intake[b]
Lung Counting	As Required	>100 µg ^{235}U or 7 nCi Total U	Recount to confirm result and perform urinalysis

a. In addition, personnel may be assigned a special frequency if deemed necessary by HP.

b. When intake is confirmed to be > 1 mg uranium, an investigation is performed to identify the source of the exposure, assess the impact, and if practical, a means to prevent reoccurrence.

Table 4.7-4
DAC and Airborne Radioactivity Posting Levels

NUCLIDE[a]	DAC[c, d]	POSTING LEVEL[b]
Gross Alpha based on Class D ^{234}U and 2 percent Class W ^{230}Th	1.0×10^{-10}	1.0×10^{-11}
Gross Alpha based on Class D ^{234}U and 8 percent Class W ^{230}Th	3.0×10^{-11}	3.0×10^{-12}
Gross Alpha based on Class W ^{230}Th	3.0×10^{-12}	3.0×10^{-13}
Gross Beta-Gamma based on Class Y ^{234}Th	6.0×10^{-8}	6.0×10^{-9}

a. All values are listed with units of µCi/mL.

b. Posting Levels are 10 percent of DAC.

c. The values above are assumed as worst case, i.e., ^{230}Th is present in each mixture at the highest concentration per category as described.

d. Area may be posted based on calculated DACs from actual airborne radioactivity concentration data.

4.8 Additional Program Elements

4.8.1 Posting and Labeling

Caution signs for Radioactive Material Areas (RMAs), ARAs, RAs, and HRAs are maintained as required by 10 CFR 20.1901, 20.1902, 20.1903, 20.1904, and 20.1905. RMAs located within a posted CCZ, CA, HCA, ARA, RA, HRA or other posted radiological area are not required to be posted as an RMA since a higher level of control is already required. In addition, as noted in Section 1.2.5 of this license application, the following exception to the applicable 10 CFR Part 20 requirements has been taken and requires an exemption:

- UF_6 feed, product, and depleted uranium cylinders, which are routinely transported inside the reservation boundary between facility locations and/or storage areas at the facility, are readily identifiable due to their size and unique construction, and are not routinely labeled as radioactive material. Qualified radiological workers constantly attend UF_6 cylinders during movement.

4.8.2 Contamination Control

4.8.2.1 Access to Restricted Areas

Restricted Areas are areas to which access is limited by the Licensee to protect individuals against undue risks from exposure to radiation and radioactive materials. Unescorted access to Restricted Areas requires the successful completion of the appropriate level of radiological worker training and, if required, a personnel dosimeter. Depending upon the type and extent (or amount) of radioactive material present, Restricted Areas are further identified as RMAs, CCZs, CAs, HCAs, ARAs, RAs, or HRAs.

Radiological control is provided by controlling access to Lead Cascade 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. Access and departure requirements are specified by procedure and/or reiterated in RWPs. Radiological posting is used to alert personnel to the presence of radiation and radioactive materials, aid in minimizing exposures, and prevent the spread of contamination. Where contamination is present, contamination controls are implemented.

Table 4.8-1 provides definitions and criteria used for posting Lead Cascade Restricted Areas.

4.8.2.2 Equipment and Personnel Monitoring

Personnel exiting areas controlled for removable contamination (CCZs and CAs) are required to monitor themselves for contamination after removing their protective clothing and prior to leaving the step-off pad area. Equipment and materials are monitored and decontaminated if required prior to removal from, or are contained and controlled as radioactive material.

4.8.2.3 Personal Protective Equipment

Protective clothing is provided for personnel entering contaminated areas. The type(s) of clothing required is consistent with the individual's work assignment and is dependent upon the type and level of contamination anticipated. With the exception of emergency evacuations, protective clothing is removed prior to exiting the BCS as specified in Radiation Worker Training, RWP, area posting, or procedures. During emergency evacuations, personnel report to designated assembly points and/or monitoring stations where protective clothing is removed and contamination monitoring is performed.

Industrial safety equipment, such as face shields, goggles, and acid suits are available. In addition full-face negative pressure respirators and full-face positive pressure respirators and other National Institute for Occupational Safety and Health and Mine Safety and Health Administration approved devices may also be utilized for respiratory protection in accordance with Section 4.6.2 of this chapter.

4.8.2.4 Release of Materials and Equipment

Materials and equipment are not released for unrestricted use unless the contamination levels are less than the levels specified in Table 4.6-1. Contamination surveys are performed on materials, equipment, and facilities to be released from radiological controls.

Use histories are used to supplement surveys of materials or equipment that have inaccessible surfaces. Use histories are summaries of the operational history of the item. Use history information includes the function, location(s) where the item was used, and other relevant evidence to assess the item's potential for internal contamination.

Bulk, aggregate materials, or waste to be released for unrestricted use or disposal is specified in the Radioactive Waste Management Program Document.

4.8.3 Radioactive Source Control

The Radioactive Source Control Program maintains administrative and physical control of sealed radioactive sources. The Source Control Program establishes source custodians and requires leak testing, accountability, and control of sealed radioactive sources.

Each sealed source containing more than 10 microcurie (μCi) of special nuclear material is tested for leakage and/or contamination at intervals not to exceed six months. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, the sealed source is not put into use until tested. (Source and byproduct material sources used at the Lead Cascade are authorized utilizing the PORTS NRC Certificate of Compliance.)

Sealed plutonium alpha sources containing 0.1 μCi or more of plutonium, when not in use, are stored in a closed container adequately designed and constructed to contain plutonium that might otherwise be released during storage. When in use, the Licensee will test the sources at least every three months using radiation detection instruments capable of detecting 0.005 μCi of alpha contamination.

Leak tests are taken from the source or from appropriate accessible surfaces of the container or from the device where the sealed source is mounted or stored where one might expect contamination to accumulate. Leak testing is conducted by HP. The test is capable of detecting the presence of 0.005 μCi or more of removable contamination, or if a plutonium source has been damaged or broken, the source will be deemed to be losing plutonium.

The Licensee will immediately withdraw the sealed source from use and repair or dispose of the source, if determined to be leaking. Within five days after determining that any source has leaked, the Licensee will file a report with the Director, Nuclear Material Safety and Safeguards, describing the source, test results, extent of contamination, apparent or suspected cause of source failure, and corrective action taken. A copy of the report will be sent to the NRC Regional Administrator, Region II.

The periodic leak test does not apply to sealed sources that are stored and not being used. The sources excepted from this test will be tested for leakage prior to any use or transfer to another person unless they have been leak tested within six months, or three months for a sealed plutonium source, prior to the date of use or transfer.

4.8.4 Radiation Protection Instrumentation

Radiation dose rate and contamination survey instruments are selected to measure the types and energies of radiation encountered with gas centrifuge enrichment operations. As such, there is little need for a wide range of instruments. However, survey instruments capable of supporting radiography operations are maintained in inventory.

The primary complement of instrumentation includes alpha/beta count rate and scaler instrumentation plus ion chambers used to evaluate shallow dose and deep dose equivalent readings. Table 4.8-2 describes typical instrumentation available to support the operation of the Lead Cascade.

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 American National Standards Institute (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 prior to noon that day, 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. The Licensee 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 as required by 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 Lead Cascade incidents are described in Section 11.6 of this license application.

Table 4.8-1
Posting Criteria

AREA	CRITERIA	POSTING
Radiation Area measured at 30 cm	>0.005 rem/hr but ≤ 0.1 rem/hr	"CAUTION, RADIATION AREA" "TLD and RWP Required for Entry"
High Radiation Area measured at 30 cm	>0.1 rem/hour but ≤ 1.0 rem/hr	"CAUTION, HIGH RADIATION AREA" "TLD, Supplemental Dosimeter and RWP Required for Entry"
High Radiation Area measured at 30 cm	>1.0 rem/hr	"DANGER, HIGH RADIATION AREA" "TLD, Supplemental Dosimeter and RWP Required for Entry"
Very High Radiation Area measured at 1 m	> 500 rads/hr	"GRAVE DANGER, VERY HIGH RADIATION AREA" "Special Controls Required for Entry" "Contact PSS Before Entry"
Contamination (Removable)	Levels > 1 time but ≤ 100 times Table 4.6-1 values	"CAUTION, CONTAMINATION AREA" "RWP Required for Entry"
High Contamination (Removable)	Levels >100 Times Table 4.6-1 values	"CAUTION, HIGH CONTAMINATION AREA" "RWP Required for Entry"
Fixed Contamination[a]	Removable Contamination < Table 4.6-1 levels and total contamination levels > Table 4.6-1 column 3 values	"CAUTION, FIXED CONTAMINATION AREA"
Airborne Radioactivity Area	Levels 0.1 Times Table 4.7-4 DAC values	"CAUTION, AIRBORNE RADIOACTIVITY AREA" or "CAUTION AIRBORNE RADIOACTIVITY AREA" "Respiratory Protection Required"
Contamination Control Zone	Levels normally less than Table 4.6-1 removable column values with potential to exceed Table 4.6-1 removable column values	"CAUTION, CONTAMINATION CONTROL ZONE"
Radioactive Material Area or Radioactive Material Storage Area[b]	An amount of radioactive material used or stored exceeding 10 times the quantity of such material specified in 10 CFR Part 20, Appendix C	"CAUTION" "Radioactive Material Area" or "Radioactive Material Storage Area"

[a]If the area has been sealed with contrasting fixatives or alternative methods and labeled in accordance with methods approved by the RPM, the area is exempt from posting as a Fixed Contamination Area.

[b]Areas posted as a Contamination Control Zone, Contamination Area, High Contamination Area, Airborne Radioactivity Area, Radiation Area, High Radiation Area, or Very High Radiation Area need not be posted as Radioactive Materials Area.

Table 4.8-1 (continued)
Posting Criteria

Definitions

Airborne Radioactivity Area (ARA) — Any area where the measured concentration of airborne radioactivity, above natural background, may be reasonably expected to exceed either: (1) 10 percent of the DAC sampled over 8 hours, (2) a peak concentration of 1 DAC sampled over no more than 1 hour, or (3) soluble uranium concentration exceeds 50 $\mu g/m^3$ averaged over 8 hours.

Contamination Area (CA) — An area where transferable contamination levels are greater than the release limits stated in Table 4.6-1, but less than or equal to 100 times those limits.

Contamination Control Zone (CCZ) — An area where transferable contamination levels are less than the release limits stated in Table 4.6-1. CCZs are essentially buffer zones established where discrete areas of contamination may be occasionally encountered as a result of facility size.

Fixed Contamination Area (FCA) — An area containing radioactive material that cannot be readily removed from surfaces by nondestructive means, such as casual contact, wiping, brushing, or washing.

High Contamination Area (HCA) — An area where transferable contamination levels are greater than 100 times the limits stated in Table 4.6-1.

High Radiation Area (HRA) — An area, accessible to personnel, in which radiation levels could result in a person receiving a dose equivalent in excess of 0.1 rem Deep Dose Equivalent (DDE) in 1 hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

Radiation Area (RA) — An area, accessible to personnel, in which radiation levels could result in a person receiving a dose equivalent in excess of 0.005 rem DDE in 1 hour at 30 cm from the source or from any surface that the radiation penetrates.

Radioactive Material Area (RMA) — An area or structure where radioactive material is used, handled or stored.

Restricted Area — An area, to which access is limited for the purpose of protecting individuals against undue risk from exposure to radiation and radioactive materials.

Very High Radiation Area (VHRA) — An area, accessible to personnel, in which radiation levels could result in a person receiving an absorbed dose in excess of 500 rads in one hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates.

Table 4.8-2
Radiological Protection Instrumentation and Capabilities

Instrument	Manufacturer	Use	Detection Limit
LB5100	Tennelec	Air sample counting and Removable contamination sample counting	alpha - 4 pCi beta-gamma - 8 pCi alpha- 20 dpm/100 cm^2 beta-gamma - 40 dpm/100 cm^2
LB1043AS	Berthold	Personnel contamination monitoring	5,000 dpm/100 cm^2 total contamination[a]
PCM2	Eberline	Personnel contamination monitoring	5,000 dpm/100 cm^2 total contamination
Ludlum 12 with GM probe	Ludlum	Alpha personnel contamination monitoring and removable contamination surveys	100 cpm above background[b]
Ludlum 12 with alpha scintillator	Ludlum	Beta-gamma personnel contamination monitoring and removable contamination surveys	100 cpm above background[b]
REM 500	Health Physics Instruments	Neutron Dose/Dose Rate	0.001 rem (rad)/hr - 999 rem (rad)/hr
Teletector	Eberline	Beta-gamma Dose/Dose rate	0 mR/hr - 1000 R/hr
RO2	Ludlum	Beta-gamma Dose/Dose rate	0 mR/hr - 5 R/hr

a. The Berthold Monitors are set to alarm with 95 percent confidence upon detection of less than or equal to 5,000 dpm total contamination per detector. The actual detection limits are approximately 3-sigma above background, and depends on detector size, efficiency, background, and count time.

b. Personnel are trained in Radiation Worker Training to notify HP when contamination is detected greater than 100 counts per minute (cpm) above background. The maximum acceptable background count rate is 300 cpm.

c. Minimum calibration frequency is annual or manufacturer recommendations.

The instruments listed above are used for routine operations. Additional instruments are available to support emergency response.

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

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

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

5.0 NUCLEAR CRITICALITY SAFETY

The Lead Cascade has a possession limit of 250 kilograms (kg) uranium hexafluoride (UF_6), which includes a limit of 700 grams (g) enriched ^{235}U. The specific authorized uses for each class of U.S. Nuclear Regulatory Commission (NRC)-regulated material are shown in Table 1.2-3 of this license application. Notwithstanding the small inventory of licensed material, the Licensee 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, all 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 Lead Cascade Nuclear Criticality Safety (NCS) program. The program is based on the well-established Portsmouth Gaseous Diffusion Plant (PORTS) NCS program, which has been extensively reviewed by the NRC and is implemented through written procedures.

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 concluded that there were no inadvertent nuclear criticality accident scenarios that could be identified for the Lead Cascade, due to the small quantity of ^{235}U that is present in the facility. The facility has established a threshold of 1 weight (wt.) percent or higher enriched ^{235}U and 100 g or more of ^{235}U for determining when an evaluation for NCS considerations of planned operations must be performed. A 100 g ^{235}U limit was chosen as a threshold above which a Nuclear Criticality Safety Evaluation (NCSE) is required. This 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 Lead Cascade NCS program features described in this chapter may not be required to be implemented. 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 Lead Cascade performance in establishing NCS analyses and controls for the design and operation of a uranium enrichment facility.

5.1 Management of the Nuclear Criticality Safety Program

5.1.1 Program Elements

The NCS program described in this chapter is implemented by NCS procedures. The NCS procedures address Lead Cascade personnel NCS responsibilities, adherence to Nuclear Criticality Safety Approval (NCSA) requirements, review and approval of fissile material operations, posting and labeling requirements, response to NCSA violations, and NCS training requirements. Controls and/or barriers that are relied on to prevent inadvertent criticalities may be designated as items relied

on for safety (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.

5.1.2 Program Objectives

The NCS program meets the requirements of 10 CFR Part 70. The objectives of the program include:

- Preventing an inadvertent nuclear criticality;
- Protecting against the occurrence of an identified accident sequence in the ISA Summary that could lead to an inadvertent nuclear criticality;
- Complying with the NCS performance requirements of 10 CFR 70.61;
- Establishing and maintaining NCS safety parameters and procedures;
- Establishing and maintaining NCS safety limits and NCS operating limits for IROFS;
- Conducting NCS evaluations to assure that under normal and credible abnormal conditions all nuclear processes remain subcritical, and maintain an approved margin of subcriticality for safety;
- Establishing and maintaining NCS IROFS, based on current NCS determinations;
- Providing training in emergency procedures in response to an inadvertent nuclear criticality;
- Complying with NCS BDC requirements in 10 CFR 70.64(a);
- Complying with the NCS ISA Summary requirements in 10 CFR 70.65(b); and
- Complying with the NCS ISA Summary change process requirements in 10 CFR 70.72.

5.2 Organization and Administration

5.2.1 Nuclear Criticality Safety Responsibilities

The General Manager, American Centrifuge Plant Operations has overall responsibility for NCS and approves the implementation of NCSAs. The General Manager, American Centrifuge Plant Operations assigns responsibilities and delegates commensurate authority to Lead Cascade managers/supervisors for the implementation and oversight of the NCS requirements. The managers/supervisors ensure that sufficient resources are available for implementation of NCS requirements. The Nuclear Safety Manager, who reports to the Technical Services Manager, is responsible for implementing the Lead Cascade NCS program.

The Lead Cascade organization managers are responsible for ensuring that operations involving uranium enriched to 1 wt. percent or higher ^{235}U and 100 g or more of ^{235}U are identified and evaluated for NCS prior to initiation of the operation. The organization managers or their designees are also responsible for ensuring NCS approvals are requested, and for ensuring implementation of the requirements contained in the approvals for these same operations.

Management is responsible, in their respective operations, for ensuring that personnel are made aware of the requirements and limitations established by approved NCSAs either through pre-job briefings, required reading, training, and/or procedures (based on the complexity of the change). These managers/supervisors are responsible for ensuring any new fissile material operations that do not have approved NCSAs will not be performed until the necessary approvals have been obtained. Management is responsible for ensuring that only personnel who have received and passed NCS training as specified in Lead Cascade NCS procedures will handle fissile material.

Managers/supervisors are trained in NCS and ensure all appropriate personnel receive NCS training as specified in Lead Cascade NCS procedures. This training provides personnel with the knowledge necessary to fulfill their NCS responsibilities. Section 11.3.7 of this license application discusses the NCS training program in more detail.

The fissile material operators are responsible for conducting operations in a safe manner in compliance with procedures and are required to stop operations if unsafe conditions exist.

The Nuclear Safety Manager (or designee) is responsible for the administration of the NCS program. This includes reviewing the overall effectiveness of the NCS program, ensuring that NCS staff members are placed, trained, and qualified in accordance with written procedures, and that NCSEs and NCSAs are prepared and technically reviewed by qualified NCS engineers. NCS is independent of organizations that require NCSAs.

Qualified NCS Engineers and Senior NCS Engineers are responsible for performing the following functions:

- Providing NCSAs for fissile material operations;
- Performing walk-throughs of facilities which handle fissile material and advising appropriate management of any NCS concerns;
- Participating in investigation of incidents involving NCS and in the determination of recommendations for eliminating such incidents;
- Assisting in emergency preparedness planning;
- Providing support to the Facility Safety Review Committee (FSRC); and
- Participating in the review of procedures that involve fissile material operations to ensure NCSA commitments have been effectively incorporated into operating procedures.

NCS personnel have the authority to halt any unsafe activity.

The responsibilities of Senior NCS Engineers performing technical reviews of NCSEs are specified in the NCS evaluation and approval procedure. These responsibilities include:

- Verifying that sufficient information is documented to allow independent analysis;
- Verifying that credible process upsets related to criticality safety are properly identified and evaluated;
- Verifying compliance with the double contingency principle;
- Checking for accuracy; and
- Verifying applicability of the calculational methods.

5.2.2 Nuclear Criticality Safety Staff Qualifications

Qualified NCS Engineers meet the requirements specified in the NCS qualification procedure. The minimum requirements for a qualified NCS Engineer are:

- Baccalaureate in engineering, mathematics, or related science;
- Familiarization with NCS by having a minimum of one year experience at a comparable facility;
- Completion of NCS-related training course and KENO V.a training course or equivalent;
- Performance of at least four evaluations under the direction of a Senior NCS Engineer;
- Performance of walk-through inspections under the guidance of a qualified NCS Engineer; and
- Additional training in the physics of nuclear criticality and in associated nuclear safety practices if the trainee does not have a nuclear engineering or physics background.

The Nuclear Safety Manager can modify the minimum qualified NCS Engineer qualification requirements for personnel who have worked for a minimum of three years at other facilities as a NCS Engineer.

The minimum requirements for a qualified Senior NCS Engineer are:

- Completion of the minimum requirements for a qualified NCS Engineer;
- Performance of the functions of a qualified NCS Engineer;
- Completion of one year as a qualified NCS Engineer; and
- Be approved by the Nuclear Safety Manager.

The Nuclear Safety Manager may modify the minimum Senior NCS Engineer qualification requirements for personnel who have worked for a minimum of five years at other facilities as a nuclear criticality safety engineer.

5.3 Management Measures

5.3.1 Procedure Requirements

Operations to which NCS pertains are governed by written procedures or work packages. These procedures or work packages contain the appropriate NCS controls for processing, storing, and handling fissile material. The NCSA requirements that specify employee actions are incorporated into procedures or work packages and are identified. Identifying these requirements ensures changes to these requirements are not made without review and approval by NCS. The NCSA requirements are incorporated into the appropriate procedures or work packages as required by the NCS evaluation and approval procedure.

New and modified procedures are reviewed by the appropriate safety organizations, including NCS, as specified in the procedure for procedure control. NCS reviews the procedures to verify that the appropriate NCSA requirements have been incorporated and to verify that the proposed operation complies with NCS program requirements. Section 11.4 of this license application provides more details related to the procedure development and change process.

5.3.2 Posting and Labeling Requirements

NCS limits and controls for areas, equipment, and containers are presented through the use of postings and labels as specified in approved NCSAs and procedures. Postings and labels are proposed, reviewed, and approved during the NCSA review and approval process. These limits and controls are posted on the NCS requirements signs as required by the facility NCS procedures. Approved NCSAs specify the wording for the postings. Labels are prepared in accordance with the facility NCS procedures and used as required by NCSAs. Limits and controls are printed or written in an appropriate size, and the postings and labels are placed in conspicuous locations determined by the line organization.

5.3.3 Change Control

A Configuration Management (CM) Program ensures that any change from an approved baseline configuration is managed so as to preclude inadvertent degradation of safety or safeguards. The CM Program, described in Section 11.1 of this license application, includes organization and administrative processes to ensure accurate, current design documentation that matches the facility's physical configuration while complying with applicable requirements. The CM Program applies to NCS and a change control process is utilized that helps ensure that the requirements of 10 CFR 70.72 are met, including the ISA Summary update requirements contained in 10 CFR 70.72(d)(3).

Functional and physical characteristics of operations controlled for NCS are described in NCSAs and NCSEs. These components and features that are identified in the NCSAs and NCSEs are analyzed to determine the "boundary" of the system, encompassing those items that are essential to ensure availability and reliability. The boundaries are identified on system drawings, and the configuration is verified to be as-built. These components and features are maintained in a design control document for the Lead Cascade. Each time a change is planned, the document is reviewed by the individual (e.g., design authority, systems engineer, operations manager, maintenance, etc.) planning the change to determine if the change affects an IROFS. The NCS program establishes and maintains NCS safety limits and NCS operating limits for IROFS in nuclear processes and maintains adequate management measures to ensure the availability and reliability of the IROFS.

The change control process specifies the organizations required to perform reviews of changes. The required approvals are obtained before the change is implemented. If an item is relied on for the criticality safety of an operation, it will be identified and NCS approval is required before implementing the change. NCS reviews the NCSE for this specific operation and determines if the change affects the analysis performed and the conclusions made in the NCSE. The change request will be approved by NCS only if the change does not adversely impact NCS, or once a revised NCSE has determined that the change is acceptable and meets NCS program requirements. If a change affects the ISA Summary, it is updated appropriately. In this way, modifications to controlled operations are evaluated and approved prior to implementation.

Records Management and Document Control (RMDC) is another element of CM and is described in Section 11.7 of this license application. Procedures, documents, and records control programs provide for centralized control and issuance of documents essential to the maintenance of the design history, and a repository for records to verify this maintenance. NCSAs and NCSEs are specifically included in the index of documents that are required to be controlled.

5.3.4 Operation Surveillance and Assessment

To ensure that the NCS program is properly established and implemented, walk-throughs, assessments, and audits are utilized.

NCS walk-throughs of facilities that may contain fissile material operations are performed by NCS personnel to determine the adequacy of implementation of NCS requirements and to verify that conditions have not been altered to adversely affect NCS. These walk-throughs are performed

as specified by the NCS procedure on walk-throughs. For example, a walk-through inspection can be performed in response to trend data, at the request of the operations personnel, or due to concerns raised by employees or NCS personnel. As a minimum, these walk-throughs are completed for applicable areas annually and may be performed in conjunction with the assessments discussed below.

Internal audits of the NCS program are conducted or coordinated by the Quality Assurance Manager(s) as described in Section 11.5 of this license application. These audits are performed at a frequency necessary to ensure that each operation is audited every two years. The purpose of these audits is to determine the adequacy of the overall NCS program. This includes the adequacy of the NCSEs, NCSAs, internal assessment programs, and implementation of the NCS requirements.

The results of these walk-throughs, assessments, and audits are documented and reported to appropriate management.

NCS deficiencies are reported in accordance with the requirements contained in 10 CFR Part 70, Appendix A or other appropriate reporting requirements. Incident reporting and investigation is described in Section 11.6 of this license application. The deficiency data is trended to monitor and prevent future violations. Deficiencies are grouped into categories such as organization, building, type of process, and type of deficiency. Corrective actions are taken for identified deficiencies in accordance with the Quality Assurance Program Description and the Corrective Action Program, and records of actions taken are retained in accordance with RMDC requirements described in Section 11.7 of this license application.

5.4 Methodologies and Technical Practices

5.4.1 Adherence to American National Standards Institute/American Nuclear Society Standards

The NCS program has been developed to comply with the standards American National Standards Institute (ANSI)/American Nuclear Society (ANS) ANSI/ANS-8.1-1983 and ANSI/ANS-8.19-1996, as discussed in this section.

5.4.2 Process Evaluation and Approval

Each operation involving uranium enriched to 1 wt. percent or higher ^{235}U and 100 g or more of ^{235}U is evaluated for NCS prior to initiation. The operation and related NCS requirements are documented in a NCSA. The evaluation is documented in a NCSE. The evaluation and approval process is governed by written procedures.

When an NCSA is needed for a particular operation, the organization responsible for performing the operation submits a request to evaluate the operation. The request is approved and signed by the manager of the operating group or his/her designee. The request is then submitted to NCS for analysis.

In response to the request, a NCS evaluation is performed. The NCSE is prepared to document the analyses performed as specified in the NCS evaluation and approval procedure. A hazard identification process (e.g., a "What-If" analysis) is used to identify and document potential upset conditions, or contingencies, presenting NCS concerns. Engineering judgment of the qualified NCS engineer may indicate the need for a more detailed study. For example, a hazards and operability study may be used if the operation is complex and involves multiple interacting systems that require substantial input from operations, maintenance, and other subject matter experts to identify the possible upset conditions. A contingency analysis is performed, in which the subcriticality of a process, given the occurrence of the contingency is assessed. This analysis demonstrates the double contingency principle for the proposed operation.

The double contingency principle as stated in ANSI/ANS-8.1-1983, Section 4.2.2, is: "Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." The Lead Cascade NCS program meets the double contingency principle by implementing at least one control on each of two different parameters or implementing at least two controls on one parameter. Controls include passive engineered barriers (e.g., structures, vessels, piping, etc.); active engineered features (e.g., valves, thermocouples, flow meters, etc.); reliance on the natural or credible course of events (e.g., relying on the nature of a process to keep the density of uranium oxide less than a specified fraction of theoretical); and administrative controls that require performance of human actions in accordance with approved procedures, or by other means that limit parameters within specified values. If two controls are implemented for one parameter, the violation or failure scenarios of the controls shall be independent. Application of this principle ensures that no single credible event can result in an accidental criticality or that the occurrence of events necessary to result in a criticality is not credible.

The basis for a parameter or process condition change that could lead to a criticality being unlikely must be documented in the NCSE. The basis may be an engineered feature, administrative control, the natural or credible course of events, or any combination of these or other means necessary to ensure the change is unlikely to occur. The parameters or conditions relied on and the limits must be specified in the NCSE and controlled.

Where the natural or credible course of events is relied upon in whole or in part to prevent a process condition change, the factors that influence the process must be described in sufficient detail in the NCSE as items related to NCS and programmatically controlled. For items that are established, maintained, and implemented by non-NCS programs, credit for availability and reliability is established as described in Section 11.1 of this license application without the need for additional NCS controls. For situations where the NCS-credited controls do not provide adequate assurance of availability or reliability (i.e., situations where non-NCS programmatic and physical facility changes could adversely affect the intended criticality safety function of the items relied upon for criticality safety), specific NCS controls will be established, maintained, and implemented to ensure criticality safety.

The NCS evaluation process involves a review of the proposed operation and procedures, discussions with the subject matter experts to determine the credible process upsets which need to be considered, development of the controls necessary to meet the double contingency principle, and identification of the assumptions and equipment (i.e., physical controls) needed to ensure criticality safety.

Engineering judgment of both the analyst and the technical reviewer is used to ascertain independence of events and their likelihood or credibility. The basis for this judgment is documented in the NCSEs. Depending on the complexity of the operation, analytical methods such as Fault Tree and Event Tree Analyses are used in the evaluation process to examine potential accident scenarios. As necessary, qualitative or quantitative estimates of event frequency are developed to support the determination of the likelihood of an event.

Once the NCSE is completed, a technical review of the evaluation is performed and documented. The NCS staff member who performs the technical reviews of NCS evaluations is a Senior NCS Engineer who has successfully met the requirements specified in the NCS qualification procedure or is a NCS Engineer completing the technical review under the guidance of a Senior NCS Engineer.

The NCSA is prepared based on the results of the NCSE, and it documents the conditions of approval (i.e., NCS requirements) for the operation. The conditions of approval include the process conditions that must be maintained to meet the double contingency principle or preserve the documented basis for criticality safety and restrict the modes of operation to those that have been analyzed in the NCSE. The requirements to be included in operating procedures and postings are identified.

The NCSA approval process first involves the acceptance of the NCSE and NCSA by the technical reviewer (i.e., operating organization). A review is then performed in accordance with 10 CFR 70.72 as described in Section 11.1.4 of this license application to determine whether prior NRC approval of the NCSA is required. If NRC approval is not required, the NCSE and NCSA are reviewed by the FSRC and, if acceptable, approved by the General Manager, American Centrifuge Plant Operations. Editorial changes require only the approval of the Nuclear Safety Manager. Editorial changes are defined as changes that do not change the technical basis of the NCSE/NCSA. The FSRC reviews the NCSA to ensure consistency with other NCSAs and other potentially conflicting requirements or regulations. Once approved, the NCS controls, limits, evaluation assumptions, and safety items are verified to be fully implemented in the field. The operations organization and NCS personnel perform this verification process. The documentation of this verification process is maintained as a quality record along with the NCSE.

Management of the operating organization is responsible for implementing, through training and procedures, the conditions delineated in the NCSA. Operational aids such as postings, labels, boundaries for fissile material operations, and fissile material movement guidelines are provided as specified in the NCSA. The manager/supervisor ensures postings and labels are prepared and verify

that they are properly installed as required by the NCSA. The procedures are prepared or modified to incorporate the NCSA requirements. Managers/supervisors are responsible for ensuring the employees understand the procedures and understand the NCS requirements before the work begins.

Each completed NCSA is issued as a controlled document. Completed NCSEs and NCSAs are archived and retrievable as permanent quality records in accordance with the RMDC requirements described in Section 11.7 of this license application. The NCSA/NCSE process provides assurance that operations will remain subcritical under both normal and credible abnormal conditions.

Emergencies arising from unforeseen circumstances can present the need for immediate action. If NCS expertise or guidance is needed immediately to avert the potential for a criticality accident, direction will be provided orally or in writing. Such direction can include a stop work order or other appropriate instructions. Documentation will be prepared within 48 hours after the emergency condition has been stabilized.

New operations must comply with the double contingency principle.

5.4.3 Design Philosophy and Review

Designs of new fissile material equipment and processes must be approved by NCS before implementation and will include the use of engineered controls on mass, geometry, moderation, volume, concentration, interaction, or neutron absorption, as the preferred approach over the use of administrative controls. Advantage will be taken of the nuclear and physical characteristics of process equipment and materials, provided control is exercised to maintain them.

The preferred design approach includes two goals. The first is to design equipment with NCS independent of the amount of internal moderation or fissile concentrations, the degree of interspersed moderation between units, the thickness of reflectors, the fissile material density, and the fissile material chemical form. The second is to minimize the possibility of accumulating fissile material in inaccessible locations and, where practical, to use favorable geometry for those inaccessible locations. The adherence to this approach is determined during the preparation, and technical review of the NCSE performed to support the equipment design. This preferred design approach is implemented as described in NCS procedures.

Fissile material equipment designs and modifications are reviewed to ensure that engineered controls are used to advantage. Administrative limits and controls will be implemented to satisfy the double contingency principle for those cases where the preferred design approach cannot be met. The basis for the decision where the preferred design approach is not used is documented.

5.4.4 Criticality Accident Alarm System Coverage

The requirements in 10 CFR 70.24 specify that each licensee authorized to possess special nuclear material in a quantity exceeding 700 g of ^{235}U must provide a monitoring system to detect an accidental criticality and alert personnel that the accident occurred. As shown in Table 1.2-1,

which defines the possession limits for the Lead Cascade, the facility has a limit on enriched uranium of 700 g of ^{235}U. Accordingly, a Criticality Accident Alarm System is not required in the Lead Cascade to detect an inadvertent criticality and alert personnel.

5.4.5 Technical Aspects

5.4.5.1 Application of Parameters

Moderation

Water and oil are considered to be the most efficient moderators commonly found in the Lead Cascade. 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.

Moderation control is applied to Lead Cascade equipment.

The cascade is a closed system designed to process gaseous UF_6. This closed system prevents the introduction of moderation due to wet air in-leakage. Also, because UF_6 reacts chemically with moisture (a moderator) to produce solid uranium-bearing compounds that impedes the proper operation of the cascade, the entire cascade is designed to minimize introduction of moisture.

Volume

Volume limits are used as specified in NCSAs. The bases for volume limits are provided in each NCSE prepared for those operations requiring containers. Specific details of these bases can be obtained by referring to the applicable NCSE.

Interaction

Interaction is controlled by spacing items bearing fissile material when those items could result in a criticality accident if not properly spaced. The spacing necessary to maintain a safe array of fissile material units is determined in the NCSE performed for the array. The spacing requirements are documented in the NCSA for the operation. The amount of spacing needed between items is determined based on analysis of the normal and credible abnormal process upset conditions for the particular operation. The basis for the spacing is documented in NCSEs. Other spacing requirements are applied on a case-by-case basis, depending on the results of a given NCSE.

Geometry

Geometry control is applied by limiting equipment dimensions for those systems that depend on the geometry for criticality safety. The geometry is determined in the NCSE that is performed for

each system and depends on the normal and credible abnormal process upsets conditions related to the specific system. Geometry controls are specified in the NCSAs.

Mass

Mass controls are applied on a case-by-case basis depending on the fissile material operation involved. The acceptable mass is determined based on the specific NCSE performed for the operation. The safe mass value depends on many factors including the geometry, the ^{235}U enrichment, composition, etc. The safe mass values are communicated to the operating personnel via the NCSAs.

Enrichment

Uranium-containing material in the Lead Cascade with ^{235}U enrichment less than 1 wt. percent is considered incapable of supporting a nuclear chain reaction, but interaction of such materials with materials of higher enrichment is taken into consideration in the specific NCSE for those operations which involve material enriched to greater than 1 wt. percent.

The maximum ^{235}U enrichment of UF_6 in the Lead Cascade is 10 wt. percent. Small quantities may be present outside of Lead Cascade equipment in the form of laboratory samples.

The maximum ^{235}U enrichment for each operation is established by the specific NCSE. The NCSA specifies the maximum acceptable enrichment for each operation. Credible process upset conditions that could alter the ^{235}U enrichment are also considered in the NCSEs.

Density

The density of materials used in a given operation is justified in the NCSE for the operation being considered. If the density must be controlled to maintain compliance with the double contingency principle, it will be documented in the specific NCSA for the operation.

UF_6 in the gaseous phase, at pressures and temperatures existing in the Lead Cascade equipment, is incapable of supporting a nuclear chain reaction even when intermixed with hydrogenous material (e.g., hydrogen fluoride [HF]). The basis for this conclusion is documented in the NCSE for the operation of the cascade. UF_6 in the gaseous phase in Lead Cascade equipment has low material density.

Heterogeneity

Heterogeneous configurations are considered for those operations that involve small fissile material and moderator regions. Heterogeneous groupings may occur for the handling of small sample containers; however, 10 wt. percent ^{235}U would be assumed for these samples that are handled on a safe mass basis. Using the homogeneous safe mass of 10 wt. percent ^{235}U would also be safe for heterogeneous 10 wt. percent ^{235}U because at this enrichment, the homogeneous and heterogeneous minimum critical masses are close in value.

Concentration

Concentration controls are used on a case-by-case basis. When the criticality safety of an operation depends on the concentration of fissile material, the medium is sampled twice, the samples are verified to be properly taken by a second individual, and the two samples are independently analyzed as required by the specific NCSA for the operation involved. The specific controls and details are documented in the NCSA for each operation that relies on concentration controls. A typical operating limit is 5 g ^{235}U per liter, regardless of enrichment. A concentration of 11.6 g ^{235}U per liter is considered subcritical, as recognized by ANSI/ANS-8.1. If under all postulated conditions, the concentration would always be less than 11.6 g ^{235}U per liter, the operation would be considered subcritical.

Reflection

Normal and credible abnormal reflection is considered when performing NCS evaluations. The possibility of full water reflection is considered when performing analyses. It is recognized that concrete can be a more efficient reflector than water, and its potential presence is considered.

Neutron Absorption

When neutron absorbers are used as NCS controls, the intended distributions and concentrations under both normal and credible abnormal conditions are maintained in accordance with the requirements of the applicable NCSA. These requirements are: representative sampling of the neutron absorber, sampling at a frequency based on the environment to which the neutron absorber is exposed, analyzing of samples for all material attributes for which credit is taken in the NCSE, and periodic inspections of fixed neutron absorbers to ensure adequate distribution as specified in the NCSA.

A NCS evaluation can take credit for the neutron absorption properties of the materials (1) added specifically for the purpose of absorbing neutrons, and (2) of construction, provided an allowance has been made for manufacturing and dimensional tolerances, corrosion, chemical reactions, and uncertainties in the neutron cross-sections.

5.4.5.2 Methods of Calculation

Based on ANSI/ANS-8.1-1983, Tables 1 and 6 respectively, the minimum mass required to achieve a critical configuration for fully enriched UO_2F_2 is 760 grams ^{235}U, and for 10% enrichment it is 1070 grams ^{235}U. The maximum amount of fissile material that may be possessed by the Lead Cascade is 700 grams ^{235}U according to Table 1.2-1 Lead Cascade Possession Limits. As the Lead Cascade cannot possess enough material to achieve a critical configuration, in accordance with ANS/ANSI-8.1-1993, it is not necessary to perform reactivity calculations. However, the results of historical, baseline, reactivity calculations obtained from the Gaseous Diffusion Plant, or other DOE complex sources, are used to illustrate the magnitude of the safety margin that exists between specific minimum critical parameters and actual worst case credible conditions associated with Lead Cascade operations or maintenance activities.

5.5 References

1. ANSI/ANS-8.1-1983, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*

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

3. Integrated Safety Analysis Summary for the American Centrifuge Lead Cascade Facility (U)

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

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

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

6.0 CHEMICAL PROCESS SAFETY

Lead Cascade operations require limited quantities of radioactive, hazardous, and toxic chemicals to support the basic process of uranium enrichment. Pursuant to 10 *Code of Federal Regulations* (CFR) 70.62, the Lead Cascade safety program includes process safety information to address hazardous materials. This chapter summarizes the chemical process safety program for the Lead Cascade, the integration of chemical safety with uranium enrichment operations, and the management systems used by the Lead Cascade for chemical safety. A description of the facility and process is provided in Section 1.1 and a description of the site is provided in Section 1.3 of this license application.

The Lead Cascade Chemical Process Safety Program is based on the well-established Portsmouth Gaseous Diffusion Plant (PORTS) program and is implemented through written procedures. Records for process safety compliance are retained in accordance with Records Management and Document Control (RMDC) requirements described in Section 11.7 of this license application.

The Technical Services Manager is responsible for the Lead Cascade Chemical Process Safety Program. Specific roles and responsibilities for the safety and health program, including chemical safety, environmental matters, and fire protection are identified in Chapter 2.0 of this license application. Chemical safety incorporates engineering and administrative controls to manage risk. Prevention is the preferred approach. Workers only use personal protective equipment (PPE) when it is necessary as a last line of defense.

6.1 Process Chemical Risk and Accident Sequences

A modest possession limit of 250 kilograms (kg) uranium hexafluoride (UF_6) has been established for the facility. No other hazardous chemicals are used in significant quantities in the Lead Cascade.

Due to chemical inventories that are maintained well below Threshold Quantities (TQs) set forth in the Occupational Safety and Health Administration (OSHA) Process Safety Management (PSM) Standard (29 CFR 1910.119) and the Environmental Protection Agency (EPA) Risk Management Program (RMP) Standard (40 CFR Part 68), these regulations do not apply to the Lead Cascade. Although, chemical inventories are maintained well below the TQs established in these regulations, the description of the Chemical Process Safety Program includes a description of PSM/RMP program requirements that would be applicable in the unlikely event that any chemical inventory increases to the TQ during operations.

Lead Cascade chemical safety consists of the integration of environmental, safety, and health management systems to address chemical hazards. Chemical safety controls are designed to mitigate the adverse effects of toxic materials used in the uranium enrichment process to workers, the public, and the environment. To achieve this objective, safety analyses, process hazard analyses (if required by PSM/RMP), and Industrial Hygiene and Safety (IHS) programs are utilized.

The Lead Cascade chemical safety controls are limited to non-radiological materials. Radiological materials are addressed in Chapter 4.0 of this license application. An Integrated Safety Analysis (ISA) was conducted for Lead Cascade operations that identifies potential accident sequences in the facility, designates items relied on for safety (IROFS) to either prevent such accidents or mitigate their consequences to an acceptable level, and describes management measures to provide reasonable assurance of the availability and reliability of IROFS. The analysis includes consideration of radiological and chemical consequences, as well as the toxicity of uranium. The ISA and ISA Summary are discussed in Chapter 3.0 of this license application. The details of the analysis are provided in the ISA Summary that is submitted for U.S. Nuclear Regulatory Commission (NRC) review separate from this license application.

6.2 Items Relied on for Safety and Management Measures

6.2.1 Items Relied on for Safety

Chemical process safety controls (i.e., IROFS) suitable to either prevent accidents or mitigate their consequences are identified in the ISA Summary.

6.2.2 Management Measures

Each of the management measures that help ensure the IROFS are available and reliable are briefly described in the following sections.

6.2.2.1 Procedures

6.2.2.1.1 Operating Procedures

Procedures are prepared in accordance with the requirements of a formal procedure system. The procedures program is described in Section 11.4 of this license application.

6.2.2.1.2 Safety and Health Program Procedures

Safety and health program procedures are developed, issued, and controlled in accordance with the facility Procedures and RMDC programs discussed in Sections 11.4 and 11.7 of this license application, respectively.

IHS programs used for chemical safety and implemented by safety and health program procedures include:

- Lockout/Tagout (LO/TO) Program
- Hazard Communication Program (HAZCOM)
- Confined Space Entry Program
- Safety and Health Work Permit Program
- Hot Work Permit Program
- PPE Program
- Signs/Labeling/Tagging Program
- Safety Training Programs

These safety and health programs apply to chemical safety as described in the program implementation documents.

6.2.2.2 Training

The Training Manager provides basic access training, general employee training, and technical job specific training as requested by the Lead Cascade Functional Managers.

Operations Technicians, maintenance personnel, management, and emergency response personnel have prerequisite and periodic training requirements that are necessary for initial and continued job qualification.

Personnel who operate, maintain, manage, handle, and have emergency response duties for chemicals are adequately trained for the particular chemical system or related activity. This training supplements the facility training program and occurs at the job-specific level.

Contractors (typically construction, maintenance, and service) personnel receive access training and facility-specific safety training prior to starting work. The contractor or the contractor-designated Safety and Health Officer has the contractual responsibility for internal contractor employee training. The Licensee also approves the contractor's Safety and Health Plan. If construction activities interface with chemical systems, Licensee representatives ensure appropriate job review, training, and guidance.

6.2.2.3 Maintenance and Inspection

Maintenance and inspection programs are summarized below and described in Sections 11.1 and 11.2 of this license application, and in the Quality Assurance Program Description (QAPD).

Technical Services develops maintenance and inspection requirements and criteria for chemical systems in conjunction with the specific facility maintenance organizations, the manufacturer's recommendations, and the ISA Summary. These chemical safety requirements are based on the functions of IROFS identified in the ISA Summary, PSM/RMP program requirements (if applicable), and manufacturer's recommendations for a particular chemical component/system.

6.2.2.3.1 Calibration and Inspection

Specific calibration and inspection requirements are based on operating characteristics, past operating experience, system operating environments, and manufacturer's recommendations.

Maintenance of chemical systems is performed in accordance with the facility maintenance programs that include the PSM/RMP Mechanical Integrity Program requirements for those systems covered under PSM/RMP. These facility programs are based upon calibration and inspection requirements from operational experience and characteristics of the system.

6.2.2.3.2 Maintenance Work Packages

Maintenance work packages are prepared to provide the necessary technical and safety guidance for maintenance activities as described in Section 11.2 of this license application. These work packages are applicable to chemical systems and equipment. Supporting maintenance procedures are subject to the requirements of the Procedures program described in Section 11.4 of this license application.

6.2.2.3.3 Preventive Maintenance and Quality Considerations

Manufacturers' recommendations are used as guides for preventive maintenance on specific chemical systems and equipment. If operational experiences or system characteristics indicate a need for a different preventive maintenance schedule, the preventive maintenance baseline can be changed after appropriate review. Equipment installed or maintenance services provided by contractors for chemical systems are tested and inspected by the contractor as required by the contract for that project. Lead Cascade personnel also perform independent inspection and testing, based on the graded approach to quality.

Independent overview of maintenance activities on chemical system hardware and requirements are addressed by the QAPD and Configuration Management (CM) Program, as applicable. These independent overview programs include:

- Procurement Quality Requirements
- Construction Inspection
- Testing and Pre-Operational Inspection
- Pressure Vessel Inspection
- Crane Inspection
- Pre-Operational Safety Review and Pre Start-up Safety Review Programs
- Facility Safety Review Committee (FSRC)

The pre-operational safety review process is conducted in accordance with program implementing procedures utilizing a graded approach. The scope of the safety review is determined by the FSRC that considers the specific issue and system being reviewed and the potential safety concerns present.

Deficiencies associated with maintenance activities are dispositioned in accordance with the QAPD and the Corrective Action Program, as described in Section 11.6 of this license application.

6.2.2.4 Configuration Management

The CM program is described in Section 11.1 of this license application. Technical Services, as the design authority for the Lead Cascade, administers the CM Program. The Lead Cascade CM Program includes an organizational structure and administrative processes and controls to ensure that accurate, current design documentation is maintained that matches the facility physical configuration.

6.2.2.5 Emergency Planning

The Lead Cascade Emergency Management Program is the responsibility of the Plant Services Manager. Emergency planning is performed by the GDP. The Lead Cascade Emergency Management program is described in Chapter 8.0 of this license application. The Emergency Management program description outlines the roles and responsibilities of personnel during an emergency and describes the emergency response measures, including onsite and offsite protective actions.

Licensee personnel who have emergency response assignments or duties associated with chemical safety are adequately trained to respond to chemical and operational upsets per 29 CFR 1910.120(q) requirements.

Operators, in compliance with the "See and Flee" policy, are not expected to participate in emergency response activities for chemical releases. The policy specifies that employees promptly move to a safe location, away from the immediate release area. Mitigating actions, as described by procedure, may be performed during evacuation from the immediate release area if they do not hinder safe egress. Personnel outside the immediate release area may perform mitigating actions, as described by procedure, prior to evacuation. If facility procedures direct an employee response to a minor spill, an employee can implement the facility Spill Prevention Control and Countermeasure (SPCC) Plan after "See and Flee" requirements have been accomplished and the area may be reentered. The SPCC incorporates the handling, storage, spill containment, cleanup, security, and spill reporting requirements for oils as per EPA regulations 40 CFR Part 112.

6.2.2.6 Incident Investigation

The Incident Investigations and reporting program and process, are described in Section 11.6 of this license application. Incident investigations and reporting are conducted in accordance with Lead Cascade procedures. The level of investigation is based upon severity and significance of the event, as well as the regulatory requirements involved. Any unacceptable performance deficiencies are addressed in accordance with the Lead Cascade Corrective Action Program. Documentation is retained in accordance with RMDC requirements.

Occupational injury and illness investigations related to chemical safety are part of IHS programs. Investigations are conducted in accordance with OSHA requirements.

6.2.2.7 Audits and Inspections

Formal Lead Cascade audit responsibilities are assigned to the Quality Assurance Manager(s). In addition, internal organizations have monitoring programs, assessments, and reviews as required by program implementation procedures. The Lead Cascade audit and assessment program is described in Section 11.5 of this license application and includes chemical safety.

6.2.2.8 Quality Assurance

The QAPD describes the quality assurance program elements and requirements that apply to Quality Level (QL) 1, 2, and 3 items. These quality assurance elements and requirements apply to chemical safety items classified as QL-1, -2, or -3 in a graded approach as described in the QAPD.

6.2.2.9 Human Factors

Human factors design responsibility for facility and system design in the Lead Cascade is assigned to engineering, with specific technical assistance from Industrial Safety personnel. Human factors reviews address the interface of people with processes and its impact on system operation.

6.2.2.10 Detection and Monitoring

A functional description of the chemical system is discussed in Section 1.1 of this license application. The Lead Cascade does not utilize a system for detection and monitoring of chemical releases, because of the design and operation of the Lead Cascade. In addition, the cascade area is small in area and manned on a continuous basis while UF_6 is present in the system.

6.2.2.11 Chemical Safety Control Strategy

The Lead Cascade chemical safety control strategy first requires that the chemicals used be identified and the listing of chemicals be kept current. Then the chemicals are reviewed for potential hazards. In order of decreasing risk and decreasing significance, the chemical hazards are addressed within the ISA Summary, PSM/RMP-specified process hazard analyses (if appropriate), and by the applicable IHS programs.

6.2.2.11.1 Identification and Inventory Control

Three processes are used to identify hazardous or toxic chemicals to be evaluated and controlled in the Lead Cascade. The first process identifies and inventories chemicals used in the Lead Cascade. This process ensures that chemicals used in the facility are appropriately addressed for safety. The process includes:

- Purchase requisition reviews;
- A listing of chemicals used;
- Material Safety Data Sheet (MSDS) library, upgrades, and distribution services to the site; and
- Identification of new chemicals for the review process.

The second process is the formal request for engineering services required for modifications to existing systems. The request process provides a mechanism that identifies new or revised usages of chemicals, chemical processes, and/or associated possible logistics that require engineering involvement. A request for engineering services may not be required unless physical modifications or updated engineering evaluations are needed. If any changes to hazardous chemical inventories or locations exist as a result of a request for a new, modified, or decommissioned facility, process or storage location, an appropriate chemical safety review will

be applied to address regulatory requirements. Any physical change to the facility, including inventory limits and changes of location for hazardous chemicals, are evaluated in accordance with the requirements of 10 CFR 70.72.

The third process is associated with contractors on site. When work is to be performed by contractors, a review of the contractors' Safety and Health Plan is conducted to identify the presence of hazardous and toxic materials to be brought onsite by the contractor. The contractor provides MSDSs for these chemicals and the list of chemicals is forwarded to the Lead Cascade Shift Supervisor.

6.2.2.11.2 Chemicals Addressed By Integrated Safety Analysis Summary

The ISA Summary addresses risks associated with UF_6 and its airborne release reaction product, hydrogen fluoride (HF). The ISA Summary provides an evaluation of accidents that involve the release of UF_6, including both radiological and toxicological hazards. The HF, which evolves from a UF_6 release, is one of the toxicological hazards. The analyses identify IROFS and administrative controls required for safety.

6.2.2.11.3 Chemicals Addressed by Process Safety Management and the Risk Management Program

Due to chemical inventories that are maintained well below TQs stipulated in the OSHA PSM Standard (29 CFR 1910.119) and the EPA RMP Standard (40 CFR Part 68), these regulations do not apply to the Lead Cascade.

6.2.2.11.4 Industrial Hygiene and Safety Program Managed Chemicals

Hazardous and toxic chemicals are effectively managed using IHS programs. Unlike PSM/RMP-covered processes, there is no minimum TQ. To address these hazards, the IHS program provides the necessary protective barriers and controls that enable safe use of these chemicals.

Commercial chemicals have varying toxicity and hazardous ranges and categories. Because chemicals can be used within the facilities in various manners, the IHS program applications to chemical safety are general in nature and based on industry accepted standards and regulatory requirements for controlling occupational exposures. To address the potential exposure risks associated with IHS program managed chemicals, the Lead Cascade uses chemical review programs, program procedures, and MSDSs. Implementation of these IHS programs provides employee protection from hazardous chemicals during daily operations and emergency response.

6.2.2.12 Multi-Occupancy of the PORTS Site

The Licensee leases from the U.S. Department of Energy (DOE) certain operating segments and certain support facilities of the PORTS reservation. The Lead Cascade and the GDP are separate entities for the purposes of chemical safety. Each has its own Chemical Safety Program, but share information regarding hazardous chemicals used at each location. DOE environmental restoration contractors and sub-contractors use the remaining site sectors. DOE provides information regarding any hazardous chemicals used by these "third-parties" that could impact Lead Cascade operations in accordance with existing protocols.

6.3 Requirements for New Facilities or New Processes at Existing Facilities

System design requirements adhere to the 10 CFR 70.64 chemical Baseline Design Criteria (BDC) for new facilities. Revision or modification to a chemical system is initiated via a request for engineering services that initiates the design process and includes a 10 CFR 70.72 review. For systems subject to the requirements of the PSM/RMP programs, a pre-startup safety review is performed for new or modified facilities when the change requires a change to process safety information. The pre start-up safety review is an independent review to address the readiness of the system hardware, associated hazard controls, personnel (including required training), and procedures. If the change does not require a change to process safety information, a pre-startup safety review is not performed.

6.4 References

1. 29 *Code of Federal Regulations* Part 1910.119, Process Safety Management of Highly Hazardous Chemicals

2. 40 *Code of Federal Regulations* Part 68, Chemical Accident Prevention Provisions

3. Integrated Safety Analysis Summary for the American Centrifuge Lead Cascade Facility (U)

4. NRC Information Notice No. 88-100: *Memorandum of Understanding between NRC and OSHA Relating to NRC-Licensed Facilities* (53 *Federal Regulations* 43950, October 31, 1988), December 23, 1988

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

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

7. NUREG-1601, *Chemical Process Safety at Fuel Cycle Facilities*

7.0 FIRE SAFETY

The Licensee maintains fire safety awareness among employees, controls transient ignition sources and combustibles, and maintains a readiness to extinguish or limit the consequences of a fire. This chapter provides a description of the Lead Cascade's Fire Safety Program. This chapter also provides a description of the fire protection facilities and equipment used to protect health and safety from fires in the Lead Cascade. The Plant Services Manager has the responsibility for Lead Cascade Fire Safety.

The Lead Cascade is located in facilities at the Portsmouth Gaseous Diffusion Plant (PORTS) in the former Gas Centrifuge Enrichment Plant (GCEP) buildings. PORTS has a well established and U.S. Nuclear Regulatory Commission (NRC)-accepted fire safety/protection program which is implemented through written procedures. The Lead Cascade's Fire Safety Program is based on this program. The Lead Cascade program takes advantage of the existing programmatic elements and experience. The program draws heavily on National Fire Protection Association (NFPA) codes and standards and provides an acceptable level of fire safety. The Lead Cascade relies upon the gaseous diffusion plant (GDP) to provide fire services resources, programmatic support, and perform emergency response to fire and other types of accident scenarios occurring at the Lead Cascade.

7.1 Fire Safety Management Measures

7.1.1 Program

The Lead Cascade fire protection facilities and equipment are designed to detect, contain, and suppress fires. The major physical components of the fire protection system include the water supply system, pumps, sprinkler systems, fire alarms, and other firefighting equipment. The location and operating characteristics of these components are described in Section 7.3 of this license application. Fire protection design provides for adequate protection against fires and explosions in accordance with the Baseline Design Criteria contained in 10 *Code of Federal Regulations* (CFR) 70.64(a) and the defense in depth requirements of 10 CFR 70.64(b).

The Fire Safety/Emergency Management Manager reports to the Plant Services Manager and is responsible for the Fire Safety Program including fire services and developing, executing, and implementing the program. This manager has the authority to ensure that fire safety receives appropriate priority.

An experienced fire professional is assigned as the Authority Having Jurisdiction (AHJ) with the responsibility for the interpretation and application of applicable fire codes and standards. The AHJ is a qualified fire protection professional having a bachelor's degree in engineering or a technical curriculum and at least six years applicable experience. These requirements are similar to the eligibility requirements as Member grade in the Society of Fire Protection Engineers.

For modifications to the facility the following standards apply: NFPA 10-1990, *Standard for Portable Fire Extinguishers*; NFPA 13-1989, *Standard for the Installation of Sprinkler Systems*; NFPA 15-1990, *Standard for Water Spray Fixed Systems for Fire Protection*; NFPA 24-1992, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*; and NFPA 30-2003, *Flammable and Combustible Liquids Code*. Any areas where full compliance with these standards will not be observed will be documented and justified by the AHJ. Any deviations found during future modifications will be documented and justified by the AHJ or corrective action will be taken.

NFPA 801-1998, *Standard for Fire Protection for Facilities Handling Radioactive Materials*, provides the fire protection requirements for facilities handling radioactive materials and generally references the NFPA codes and standards dealing with each specific type of equipment or program. At PORTS, the existing NRC-regulated fire protection program already has these elements included in the program. In addition, the PORTS program is based on the NFPA codes and standards as implemented by the U.S. Department of Energy (DOE) at the time of the NRC Certification under 10 CFR Part 76. Modifications since that time were based on the guidance in the NFPA codes and standards as described in USEC-02. Committing to full compliance with NFPA 801-1998 for the Lead Cascade would require significant changes to the existing underlying program and expenditure of resources, with little to no reduction in risk or benefit to public health and safety. Alternatively, the Licensee considered whether a unique Fire Safety program could be implemented for the Lead Cascade. This option was eliminated since it would cause unnecessary complication, and possible confusion, for the PORTS fire services personnel implementing the programs, especially considering the Lead Cascade's utilization of such a small fraction of the overall leased space at the PORTS**.** For any future modifications to the fire protection program, PORTS fire services will use NFPA 801-1998 and the other NFPA codes and standards for guidance.

The Facility Safety Review Committee (described in Section 2.2.1 of this license application) provides an oversight and review role of fire safety in the Lead Cascade. The membership, structure and responsibilities of this multi-discipline committee are defined in a charter. The charter of the committee includes the responsibility to review fire safety issues and to integrate changes to the facility with adequate consideration of fire safety.

7.1.2 Pre-Fire Planning

Pre-fire plans have been developed for the X-3001, X-3012, X-7725, and X-7726 Buildings. Each pre-fire plan contains information about the building, the layout, specific hazards and other information applicable to the individual facility. Fire services personnel review these pre-fire plans as part of the facility inspection.

7.1.3 Testing and Inspection

The inspection and testing of fire protection equipment is performed by or overseen by fire services personnel to help ensure that fire safety related items relied on for safety (IROFS) are available and reliable. The testing and inspection of equipment is performed in accordance with procedures that include test frequencies as defined by Fire Safety/Emergency Management Manager. These are based on the current GDP program (that is based on NFPA inspection frequencies)

modified to the special situations at the site (such as controlled access, specially trained workforce, supervised systems, on-site fire services and engineering staff, and extensive operating experience).

The major elements of the inspection program and their associated frequencies are:

- **Annually:**
  - Flow test fire pumps
  - Flow test wet pipe sprinkler systems
  - Inspect and flow test fire hydrants
  - Test manual fire alarms (pull stations)
  - Test sprinkler waterflow alarms
  - Test supervisory alarm devices including control valves, low air pressure, low temperature, and loss of power
  - Flow test pumper trucks
  - Test self-contained breathing apparatus (SCBA)
  - Operate sprinkler system control valves
  - Test special fire alarm indicators, such as heat and smoke detection systems
  - Inspect major buildings to evaluate housekeeping, check fire emergency equipment, and exit pathways
  - Hydrostatically test fire hoses on pumper trucks
  - Test local application fire suppression systems
- **Quarterly**
  - SCBA air quality checks
- **Monthly**
  - Start test fire pumps
  - Inspect wet pipe sprinkler systems risers
  - Inspect portable fire extinguishers

7.1.4 Personnel Training

Fire services personnel are trained and equipped to handle anticipated types of emergencies. Firefighter training is equivalent to the state certified firefighter training curriculum. Emergency medical response personnel meet requirements for state certification as Emergency Medical Technicians (these are typically 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 in accordance with Records Management and Document Control requirements. Training is based on national standard emergency response methodology with site-specific training on issues unique to the site.

Specific training activities include firefighting, hazardous material response, confined space rescue, emergency medical response, radiological emergencies, and rescue. Drills are conducted quarterly as part of the Emergency Plan. Training requirements of fire services personnel are described in Section 11.3 of this license application. Fire Officers have at least five years experience in fire service with management and leadership training.

An on-site emergency squad provides additional support for the fire services group. This group is on call for response to assist emergency responders at emergency scenes. Training is provided for the type of activities they may be called upon to perform.

Employees receive initial and biennial fire safety training as part of General Employee Training (GET) on emergency preparedness. This includes emergency reporting, facility evacuation, and fire extinguisher familiarization. GET is described in Section 11.3 of this license application.

7.1.5 Impairment Control

Closure of valves on the water system supplying the fire suppression systems is controlled by a written permit system. Fire services controls the valve closure permit system; therefore, fire services is notified of the impairment of fire suppression systems. Only groups authorized by Fire Safety/Emergency Management Manager have the authority to issue permits and operate fire protection valves.

This permit system provides for the notification of the facility manager and the Plant Shift Superintendent (PSS) of the reason for the impairment, the expected duration of the impairment, the person doing the valving, system restoration time, person restoring the system, and residual partial system impairment (e.g., branch line removed). Compensatory actions are initiated when building sprinkler systems are out of service. These may include suspension of hot work or other hazardous processes, personnel notifications, fire patrols, or other action necessary as determined by the Fire Safety/Emergency Management Manager. Systems taken out of service for repair are usually returned to service within an 8-hour period; actual required repairs will affect the actual time needed to complete the repair.

7.1.6 Hot Work Permits

Hot work is controlled by procedure complying with NFPA 51B-2003 *Standard for Fire Prevention During Welding, Cutting, and Other Hot Work*. A hot work permit system is used to ensure that cutting, welding, and other hot work conducted in areas not designed or approved for such processes will be done in a manner that is consistent with industry fire prevention practices. Line managers are trained on fire safety and are authorized to write hot work permits. The Fire Safety/Emergency Management Manager, or designee, is notified by the line manager prior to the initial use of a hot work permit. The permits are logged and a field surveillance of work is conducted during routine building inspections and when concerns or unusual circumstances exist.

7.2 Fire Hazards Analysis

Fire hazards for the X-3001, X-3012, X-7725, and X-7726 buildings are evaluated annually by fire protection engineering staff and are documented in a building survey. The fire hazards evaluation activity consists of two major elements: the annual building surveys and the Integrated Safety Analysis (ISA). Each element is described below.

7.2.1 Building Survey

A survey is an inspection and analysis with a focus on fire safety. These surveys provide a formal review and periodic evaluation of the occupancy and the fire protection associated with a facility.

A building survey includes the following elements:

- Identifying building construction;
- Defining fire areas;
- Evaluating fire cutoffs or barriers;
- Determining exposures to the structure or facility;
- Describing building function including occupancy classification;
- Assessing ordinary building hazards, such as ventilation and heating systems and combustibles;
- Discussing processes including any special equipment or special operation;
- Assessing special hazards such as flammable liquid processes, high piled stock, and classified electrical installation;

- Reviewing fire protection and installed detection equipment, as well as special features of fire protection in the building;

- Developing a list of issues or recommendations for the facility manager regarding fire protection issues and tracking to resolution; and

- Reviewing emergency egress paths.

7.2.2 Integrated Safety Analysis

An ISA for operation of the Lead Cascade was conducted in accordance with the guidance provided in NUREG-1513, *Integrated Safety Analysis Guidance Document* and the requirements of 10 CFR 70.62(c). An ISA Summary was prepared.

The ISA contains these elements:

- 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 a release regardless of the initiator.

7.2.3 Hazards Evaluation

The hazards evaluation performed in this manner is consistent with the approach taken by the PORTS fire protection program in the past. Historically, evaluations have been effective in identifying issues and problems, thus facilitating the continued success of the fire protection program.

The fire protection building surveys provide the same general information as a formal Fire Hazards Analysis (FHA), with the exception of the accident analysis. The accident analysis, which fully evaluates a suite of fire scenarios, is performed as part of the Lead Cascade ISA. Accordingly, a stand-alone document called a FHA has not been prepared for the Lead Cascade; however, the elements have been addressed. The hazards evaluation approach described herein is adequate to ensure that appropriate fire hazards have been assessed, consequences evaluated, and that adequate protection to employees and the public is provided.

Review of the emergency egress paths for the facility is accomplished using the intent of NFPA 101-1991, *Life Safety Code*, as guidance. The buildings do not comply with the travel distances due to the size of the buildings. Exit arrangements are adequate because of the low occupancy levels, large number of exits, and fixed fire suppression systems in the buildings.

7.3 Facility Design

The major Lead Cascade buildings are constructed of heavy unprotected steel frame, concrete floors, insulated metal panel exterior walls, and a built up roofing and/or spray applied polyurethane, silicone material on a metal deck roof. Each building is considered a single fire area; sprinkler coverage is provided. The sprinkler and water systems are described below. There are no water-exclusion areas in the Lead Cascade. Combustible loading is low and the fire hazards are limited to normal industrial activities. Any exceptions to this are identified in the building survey report or by the facility manager. These include such things as electrical switchgear and transformers, and maintenance activities. Lightning protection is installed on the major buildings.

Firewater runoff to the environment is minimized by use of a Liquid Effluent Collection System that is employed to collect liquid discharges from the Lead Cascade buildings. Underground storage tanks outside of the buildings collect wastewater from within the buildings. There is a network of underground floor drains that collect the wastewater and drain it by gravity flow into these underground storage tanks. The tanks must be manually pumped, as no drains are installed in the tanks.

There are no major fire potentials related to the enrichment process. Fire hazards are typical industrial hazards, including construction and maintenance, incidental use of chemicals and flammable liquids and gases, and energized electrical equipment, in the process buildings. Accident potentials are discussed in the ISA.

The X-3001 and X-7725 Buildings are each considered single fire areas. Building separation is used as a method of limiting fire spread. The buildings are connected by the Transfer Corridor (X-7727H) of the same construction and have a fire sprinkler system.

Combustible storage in the process buildings is considered as part of the hazard evaluation described in Section 7.2 of this license application. The sprinkler system protects the buildings from fire and prevents structural collapse.

There are no significant quantities of flammable liquids or gases used in the enrichment process. The incidental use of these liquids and gases, primarily for construction, maintenance and support activities, is controlled using the guidance of NFPA 30-2003, *Flammable and Combustible Liquids Code* and NFPA 55-2005, *Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks*.

Hot work operations may be involved during construction and maintenance activities associated with the cascade. These hot work operations are covered by a permit system. This includes pre-job inspection, stationing a fire watch during the hot work, and post-job fire watch to prevent delayed ignition of any combustibles. Personnel performing fire watches receive both classroom and hands-on fire extinguisher training.

7.3.1 X-3001 Process Building

Fire suppression for the X-3001 Process Building is provided by wet pipe sprinkler systems. The systems are hydraulically designed and installed to meet or exceed the NFPA recommended

sprinkler density for Ordinary Hazard Group 1 occupancies. The systems consist of sprinklers located at the ceilings/roof level and in other areas where needed. The sprinkler heads are supplied by piping fed from a riser connected to the firewater distribution system.

7.3.2 X-7725 Recycle/Assembly Building and X-7726 Centrifuge Training and Test Facility

Fire suppression for the X-7725 Recycle/Assembly Building and X-7726 Centrifuge Training and Test Facility are provided by wet pipe sprinkler systems. The systems are hydraulically designed and installed to meet or exceed the NFPA recommended sprinkler density for Ordinary Hazard Group 1 occupancies and storage occupancies. The systems consists of sprinklers located at the ceilings/roof level and in other areas where needed. The sprinkler heads are supplied by piping fed from a riser connected to the firewater distribution system.

7.3.3 X-7727H Transfer Corridor

Fire suppression for the X-7727H Transfer Corridor is provided by wet pipe sprinkler systems. The systems are hydraulically designed and installed to meet or exceed the NFPA recommended sprinkler density for Ordinary Hazard Group 1 occupancies. The systems consists of sprinklers located at the ceilings/roof level and in other areas where needed. The sprinkler heads are supplied by piping fed from a riser connected to the firewater distribution system.

7.3.4 Alarms

The sprinkler systems are connected to the GCEP Fire Alarm system. This system was installed to meet the intent of NFPA 72-2002, *National Fire Alarm Code*. The system alarms include sprinkler water flow alarms from the sprinkler systems and manual pull stations located in the X-3001 and X-7725 facilities. Alarms are received in the alarm receiving area in X-1007 Fire Station. Alarm announcement is not local, but building evacuation can be manually initiated from the X-3012 Area Control Room (ACR) and, as a back-up, the X-300 Plant Control Facility.

7.4 Process Fire Safety

A number of the release scenarios evaluated in the ISA have fire as the initiating event. The ISA addresses risks associated with UF_6 and its airborne release reaction product, hydrogen fluoride (HF). The analyses identify IROFS and related management measures.

UF_6 is the primary hazardous material in the Lead Cascade and the ISA provides an evaluation of accidents that involve the release of UF_6, including both radiological and toxicological hazards. The HF, which evolves from a UF_6 release, is considered as one of the toxicological hazards from a UF_6 release and is also addressed in the ISA.

7.5 Fire Protection and Emergency Response

7.5.1 Fire Protection Engineering

Fire protection engineering support is available, if necessary, to evaluate fire hazards; review changes to maintenance and process systems; and provide in-house consultation. They also perform the building surveys as described in Section 7.2 of this license application.

Fire protection engineers assist in the development of project design criteria, perform design review, and conduct routine engineering consultation as necessary. Fire protection engineering is part of project design teams and routinely reviews project design packages to ensure applicable fire safety issues are addressed. These issues may include construction, egress, facility protection, separation of fire areas, detection systems, and special hazard protection. Fire protection engineers are either graduates of a technical program or have at least six years experience in fire protection work.

All reported fires are investigated using a graded approach. This includes investigations by Fire Officers, engineers, or by multidiscipline teams as warranted. Results of investigations are considered for distribution throughout Licensee operations to prevent future reoccurrences. Details of incident investigation in the Lead Cascade are described in Section 11.6 of this license application.

7.5.2 Fixed Fire Suppression and Fire Detection Systems

The fire alarm system monitors fire alarms in important Lead Cascade buildings and structures. A listing of these facilities is maintained by fire services. Alarms caused by non-fire conditions, such as spurious water flow alarms from pressure surges, are reviewed by fire services personnel and identified for maintenance as needed. The system includes alarm notification to the X-1007 Fire Station and, as a backup, the X-300 Plant Control Facility. The alarm room at the X-1007 Fire Station is manned by fire services personnel. During some emergency situations, when the alarm room operator is needed at the incident, the alarm receiving responsibility is transferred to X-300 where backup alarm receiving equipment is monitored. This backup capability includes water flow alarms from the sprinkler systems and manual pull stations located throughout the facilities. This process provides for prompt dispatch of emergency response personnel to investigate and resolve the alarm condition.

Manual pull stations are located throughout the facilities. Operation of a pull station initiates an alarm at the central alarm receiving location X-1007 Fire Station. This alarm is not announced locally. The Lead Cascade does not have local evacuation stations, but does have evacuation alarms that are initiated from the X-3012 ACR.

Fixed automatic fire suppression systems provide the primary means of detection, control, and suppression of fires at the Lead Cascade. These systems, primarily sprinkler systems, are installed in the process-related facilities. These fixed fire suppression systems are inspected, tested, and maintained on a regular basis in accordance with approved procedures.

A reliable fire water system with water storage, pumps, and underground piping is provided. This is a looped gridded system, intended to provide minimum outage potential. Fire pumps with water supplies are split (located on opposite sides of the site) to provide maximum reliability. It would require multiple failures to render the fire water system inoperable. Sectional valves are provided throughout the system to permit isolation in the event of a pipe break, and pumping capacities are split to provide greater reliability and redundancy.

The on-site fire services group provides emergency response. Fire alarms are not transmitted off-site to area Fire Departments. The site currently has mutual aid agreements with other Fire Departments in the area. As part of the agreements, they participate in periodic exercises held by Emergency Management. Fire services occasionally responds to off-site departments' requests for assistance (typically a few times per year).

7.5.3 Mobile and Portable Equipment

Mobile fire equipment is provided and maintained on-site to support firefighting activities and to back-up the fixed fire suppression systems. This equipment is manned by Fire Services personnel and includes a minimum of one 1,000-gpm pumper, one truck with hazardous material response capability, radiological, and rescue equipment, and one ambulance. This equipment is typically housed indoors and is equipped with the necessary hose, nozzles, breathing apparatus, meters, detection equipment, rescue equipment, and other related equipment.

SCBAs are provided for use by trained personnel in connection with emergency activities, including firefighting. Breathing air used in SCBAs meets a minimum quality of Grade D.

Portable fire extinguishers are available throughout the facility, including the process areas. Size, selection, and distribution of extinguishers is determined using NFPA 10-1990, *Standard for Portable Fire Extinguishers*, as the guidance document.

Using NFPA 17-2002, *Standard for Dry Chemical Extinguishing Systems*, as guidance, local application fire suppression systems are installed on process equipment, as needed.

7.5.4 Emergency Response

The on-site fire services group primarily handles fire emergency response. This is a fully staffed fire department, with line officers and firefighters. Emergency response to the entire site is provided by this group under the incident command of the PSS or their designee. Response capability of the group includes fire, rescue, emergency medical, process problems, spills, and confined space rescue.

The Fire Safety program is staffed with fire protection engineers and Fire Officers. The fire protection engineers are responsible for design review, specification requirements, building surveys, support to other groups, and some fire investigations. The Fire Officers are responsible for daily management of the fire fighters, including the testing program, training, alarm receiving, and emergency response.

Fire services personnel are on duty 24-hours per day to provide emergency response services and redundant firefighting capability to back-up the automatic fire suppression systems installed in the Lead Cascade. The normal minimum scheduled shift staffing is six fire services personnel. Minimum staffing for emergency response is four fire services personnel, which enables primary entry with sufficient backup. In addition to their response functions, fire services personnel are responsible for the testing and inspection of fire protection systems and related equipment.

7.6 References

1. Integrated Safety Analysis Summary for the American Centrifuge Lead Cascade Facility (U)

2. NFPA 10-1990, *Standard for Portable Fire Extinguishers*

3. NFPA 13-1989, *Standard for the Installation of Sprinkler Systems*

4. NFPA 15-1990, *Standard for Water Spray Fixed Systems for Fire Protection*

5. NFPA 17-2002, *Standard for Dry Chemical Extinguishing Systems*

6. NFPA 24-1992, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*

7. NFPA 30-2003, *Flammable and Combustible Liquids Code*

8. NFPA 51B-2003, *Standard for Fire Prevention During Welding, Cutting, and Other Hot Work*

9. NFPA 55-2005, *Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks*

10. NFPA 72-2002, *National Fire Alarm Code*

11. NFPA 101-1991, *Life Safety Code*

12. NFPA 801-1998, *Standard for Fire Protection Facilities Handling Radioactive Materials*

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

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

15. USEC-02, Application for United States Nuclear Regulatory Commission Certification, Portsmouth Gaseous Diffusion Plant, Safety Analysis Report

8.0 EMERGENCY MANAGEMENT

8.1 Emergency Plan

The Lead Cascade is located in a leased area of the U.S. Department of Energy's (DOE) Portsmouth Gaseous Diffusion Plant (PORTS) reservation, adjacent to the gaseous diffusion plant (GDP) as shown in Figure 8.1-1 (located in Appendix B of this license application). PORTS has implemented a U.S. Nuclear Regulatory Commission (NRC)-accepted Emergency Plan (Plan) in accordance with the requirements of 10 *Code of Federal Regulations* (CFR) 76.91. The Plan is mature and effectively implemented by Emergency Plan Implementing Procedures (EPIPs). The EPIPs address generic requirements for responses to incidents involving hazardous chemicals, radioactive materials, natural phenomena, and other adverse conditions throughout the PORTS reservation. The Plan helps to ensure that personnel are adequately prepared for accidents and other emergencies involving the potential release of radioactive materials and that prompt, orderly, and effective response actions are taken to mitigate the consequences of such accidents and emergencies and protect the health and safety of the public and workers at the plant. The Fire Safety/Emergency Management Manager reports to the Plant Services Manager and is responsible for Lead Cascade Emergency Management.

The NRC notice promulgating 10 CFR Part 76 confirmed that the emergency planning requirements set forth in 10 CFR Part 70 for other nuclear fuel cycle facilities are appropriate for the gaseous diffusion plants and that the requirements in 10 CFR 76.91 are based upon the emergency planning provisions in 10 CFR Part 70. The Emergency Plan design adheres to the 10 CFR 70.64 Emergency Capability Baseline Design Criteria. Accordingly, the PORTS Emergency Plan format is generally based upon NRC Regulatory Guide 3.67, *Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities* (January 1992). Additionally, the emergency action levels have been developed using examples provided in this Regulatory Guide. Based on these factors, and to ensure a consistent response by emergency response personnel, the existing PORTS Emergency Plan and the accompanying EPIPs are utilized to meet the 10 CFR 70.22(i)(1)(ii) and (i)(3) requirements for an Emergency Plan for the Lead Cascade.

A general map covering a radius of approximately 10 miles from the PORTS reservation is provided in Figure 8.1-2.

8.1.1 Emergency Plan Summary

The EPIPs implement each section of the Plan, but are not included in the Plan itself. Rather, the Plan includes a general description of the procedures that are followed in connection with each activity to demonstrate that appropriate actions can and will be taken to mitigate accident consequences and to protect the health and safety of the public and plant personnel in the event of an emergency.

The Plan provides an overall description of the comprehensive site-wide emergency preparedness program and is based in large measure on the emergency preparedness policies, procedures, and practices that have been successfully used on the PORTS reservation. This program was established to manage and respond in a consistent and integrated way to accidents or other emergency situations that may occur at the site. The structure of this program is

intended to ensure that the consequences of emergencies are promptly mitigated and that the health and safety of the public, personnel throughout the reservation, and plant personnel are protected, regardless of the cause or nature of the emergency. Therefore, the Plan addresses both radiological and non-radiological emergencies.

The scenarios addressed in the Plan include accidents involving radioactive materials, non-radioactive materials, and chemicals; fires; natural phenomena such as earthquakes and tornadoes; and security-related emergencies. The scenarios in the Plan include a large uranium hexafluoride (UF_6) release, which far exceeds, and fully bounds a release of the entire inventory of UF_6 from the Lead Cascade.

The Plan includes a general description of leased property, the DOE reservation, and the surrounding area. It identifies the types of accidents and the emergencies for which protective actions may be needed and describes the manner in which accidents are detected and classified. The Plan also contains a description of the policies and procedures that are followed for the notification of and communication with plant personnel, DOE reservation personnel, local governments, and regulatory agencies in the event of an emergency, and for the coordination of the emergency response activities of both on-site and off-site response organizations. The Plan provides a description of the responsibilities of the key individuals and organizations involved in emergency response activities and the manner in which the consequences of an emergency are mitigated and assessed. The Plan also includes separate sections and subsections addressing the establishment and maintenance of emergency response equipment, facilities, and capabilities, the training and exercises that are conducted to maintain and enhance emergency preparedness, the manner in which plant equipment and systems are restored to a safe condition after an accident and all other topics required under 10 CFR 76.91. These topics are consistent with those required by 10 CFR 70.22 (i)(3).

The Plan also confirms that the Licensee has met its responsibilities under the *Emergency Planning and Community Right-to-Know Act* of 1986 as required by 10 CFR 70.22(i)(3)(xiii). The Plan is maintained and updated by the United States Enrichment Corporation (Corporation). In accordance with 10 CFR 76.91(o) and 10 CFR 70.32(i), the Corporation may change the Plan without receiving prior NRC approval, providing the change does not decrease the effectiveness of the Plan and the NRC and affected off-site response organizations are provided with copies of any changes to the Plan within six months of the change.

In summary, the Plan is the master document addressing the emergency preparedness program and the policies, procedures, and actions that will be implemented in any emergency arising from activities on the PORTS reservation or from outside sources affecting personnel working on the site to mitigate the consequences of the emergency and protect the health and safety of the public, personnel on the DOE reservation, and gaseous diffusion plant and Lead Cascade workers.

8.1.2 Lead Cascade Facilities

The Lead Cascade occupies a very small fraction of the area of the PORTS reservation (Figure 8.1-1) and the Lead Cascade has a modest possession limit of 250 kilograms (kg) UF_6.

The type, quantity, and form of NRC-regulated source material, byproduct material, and special nuclear materials are shown in Table 1.2-1. A topographical map of the reservation is shown in Figure 8.1-3. An aerial photo is provided in Figure 8.1-4 (located in Appendix B of this license application).

The Lead Cascade enriches normal UF_6, which contains approximately 0.711 weight (wt.) percent ^{235}U, up to 10 wt. percent ^{235}U. The Lead Cascade is operated on recycle where the enriched product stream is recombined with the depleted stream prior to being re-fed to the cascade. No product withdrawals are made from the Lead Cascade other than the samples that are taken for laboratory analysis. The specific authorized uses for each class of NRC-regulated material are shown in Table 1.2-2.

Due to the small quantity of licensed material, the consequences of any accident postulated in the Integrated Safety Analysis (ISA) would be small when compared to postulated accidents at the GDP. The postulated accidents addressed in the PORTS Emergency Plan envelope and bound any conceivable accident in the Lead Cascade.

The Lead Cascade is described in Chapter 1.0, the organization is described in Chapter 2.0, and the credible accident scenarios have been evaluated in Chapter 3.0 of this license application (ISA Summary).

UF_6 leak detection instrumentation and criticality accident detection instrumentation are not utilized in the Lead Cascade. The Lead Cascade is small in area and manned on a continuous basis while UF_6 is present in the system. Abnormal operating conditions and accidents are identified by human observation, based on installed instrumentation and routine operator rounds of the facility.

8.1.3 Areas Not Addressed by Existing Emergency Plan

The existing Emergency Plan provides a comprehensive response to incidents involving hazardous chemicals, radioactive materials, natural phenomena, and other adverse conditions throughout the PORTS reservation. While no new chemicals not in use at PORTS are utilized in the Lead Cascade, there are a few topics specific to the Lead Cascade and gas centrifuge technology that are not currently addressed. These topical areas are described in the following sections.

8.1.3.1 Vent System

The purge vacuum (PV) and evacuation vacuum (EV) systems are utilized to produce the required vacuum in the gas centrifuge machines. Effluent from the systems discharge through a vent to the atmosphere. The systems are described in Section 1.1.2.7.1 and airborne effluent monitoring is discussed in Sections 9.2.1.2.1 and 9.2.2.1.3 of this license application. The PV and EV systems share activated alumina traps and the X-3001 Process Building vent. The vent stack has a height of approximately 23 meters. Four chemical traps, and an exhaust gas analyzer and gas flow monitoring capability are provided as part of the system. The efficiency of the activated alumina traps ranges from 0 to 95 percent, depending on concentration of UF_6 in the effluent stream. The flow rate is expected be comparable to the existing vents utilized at the

GDP. The maximum anticipated gaseous effluent from the Lead Cascade during normal operations is estimated to be 1.0 millicurie (mCi) of uranium over a week, or up to 0.052 curie (Ci) per year. Effluent levels are discussed in Section 9.2.2.1.1 of this license application.

8.1.3.2 Accident Sequences

The cascade feed cylinder is located in the X-3001 Process Building. The feed cylinder provides the principle source for nearly all of the accident scenarios in the Lead Cascade. The most severe accident sequence involves the release of the entire 250 kg inventory of UF_6. There are no offsite radiological consequences from the Lead Cascade accident sequences; rather the sequences are predominately concerned with related chemical consequences. Details regarding the projected doses and toxic substance concentrations are provided in the ISA Summary, along with measures credited in preventing and mitigating the events.

8.1.3.3 PORTS Emergency Plan Revision

The PORTS Emergency Plan and the EPIPs were revised to address the topics that are unique to the gas centrifuge process, including classification requirements. Specifically, the underlined information below was added:

- Section 5.0 Emergency Response Measures

  Emergency measures must be taken in response to an emergency. Upon recognizing that an emergency exists, the Emergency Response Organization (ERO) is activated. Once activation has taken place, assessments of the condition are made, corrective and protective actions are taken, and aid to affected persons is administered as required.

  After becoming aware that an emergency exists, the Plant Shift Superintendent does the following:

  - Takes actions to ensure the safety of plant personnel and the general public,
  - Takes actions to ensure safe operation/activities of the plant,
  - Classifies the emergency and makes the required notifications,
  - Takes actions to ensure that safeguards and security measures are maintained,
  - Takes actions to ensure that material control and accountability measures are maintained,
  - Performs assessment actions,
  - Performs other emergency actions as appropriate, and

- For Lead Cascade emergencies, takes actions to ensure that items relied on for safety are assessed for collateral damage.

- Section 7.2.3 Offsite Emergency Management Training

  Training is offered biennially by letter of invitation to non-licensee emergency support organizations that may be called upon to respond to emergencies at the plant, including the Lead Cascade. These agencies include local fire, law enforcement, ambulance, and hospital services. Personnel from other plant groups such as Training, Health Physics, Security, Operations, and Fire Services provide assistance as needed. This training includes the following topics and performance objectives as a minimum:

  - Site-specific information on hazards, onsite and offsite protective actions, and emergency response from personnel or organizations augmenting the ERO;

  - Orientation tours of the PORTS reservation; and

  - Information briefings for the news media on operational emergencies, site-specific hazards and responses, site points of contact, and procedures for the release of information in the event of an emergency.

- Section 9.1 Recovery

  The nature and extent of the emergency determines what recovery operations are required and the extent of the recovery organization that must be formed. A recovery plan must be flexible enough to adapt to the existing conditions. It is not possible to anticipate in advance all of the conditions that may be encountered as a result of the emergency. General principles addressed in this section serve as a guide for developing a flexible plan of action.

  Recovery includes those actions necessary to return an incident site and the surrounding environment to pre-emergency conditions to the maximum extent practical. Specific recovery plans are developed in accordance with the applicable EPIP.

  The DOE site manager is responsible for ensuring the adequacy and appropriateness of recovery operations involving non-leased portions of the facility. The General Manager, American Centrifuge Plant Operations is responsible for ensuring the adequacy and appropriateness of recovery operations involving the Lead Cascade.

The revisions to the PORTS Emergency Plan and EPIPs encompassing these topics were implemented prior to beginning operation of the Lead Cascade.

The Licensee performed additional reviews of the Emergency Plan and incorporated necessary additional specific emergency management aspects of the Lead Cascade into the existing Emergency Plan, including adding any hazardous chemicals associated with the Lead Cascade to the existing list of chemicals, with clear indication that the hazardous chemicals pertain to the Lead Cascade.

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 8.1-1
Lead Cascade Location

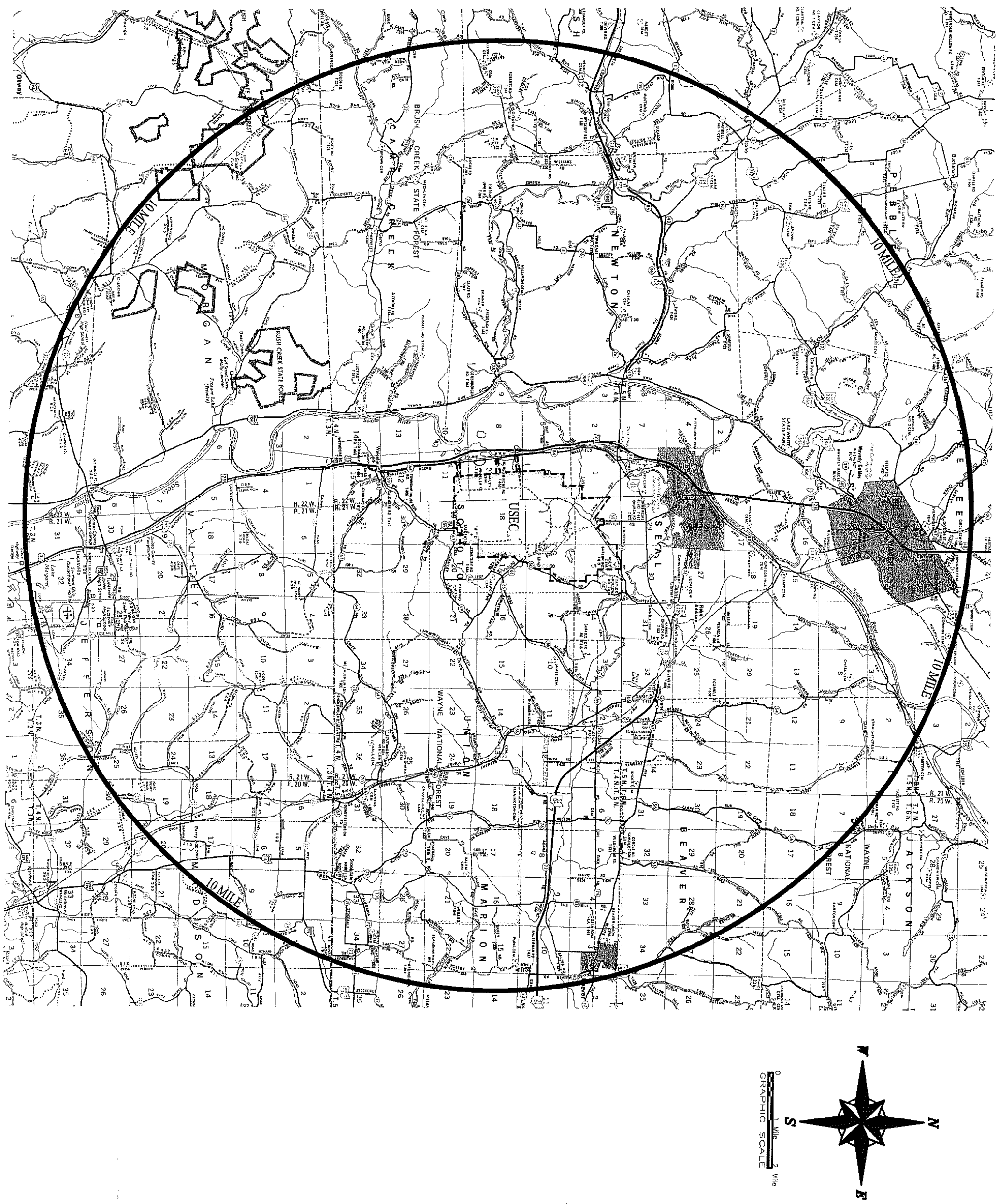

Figure 8.1-2
General Map Covering a Radius of Approximately 10 Miles from the PORTS Reservation

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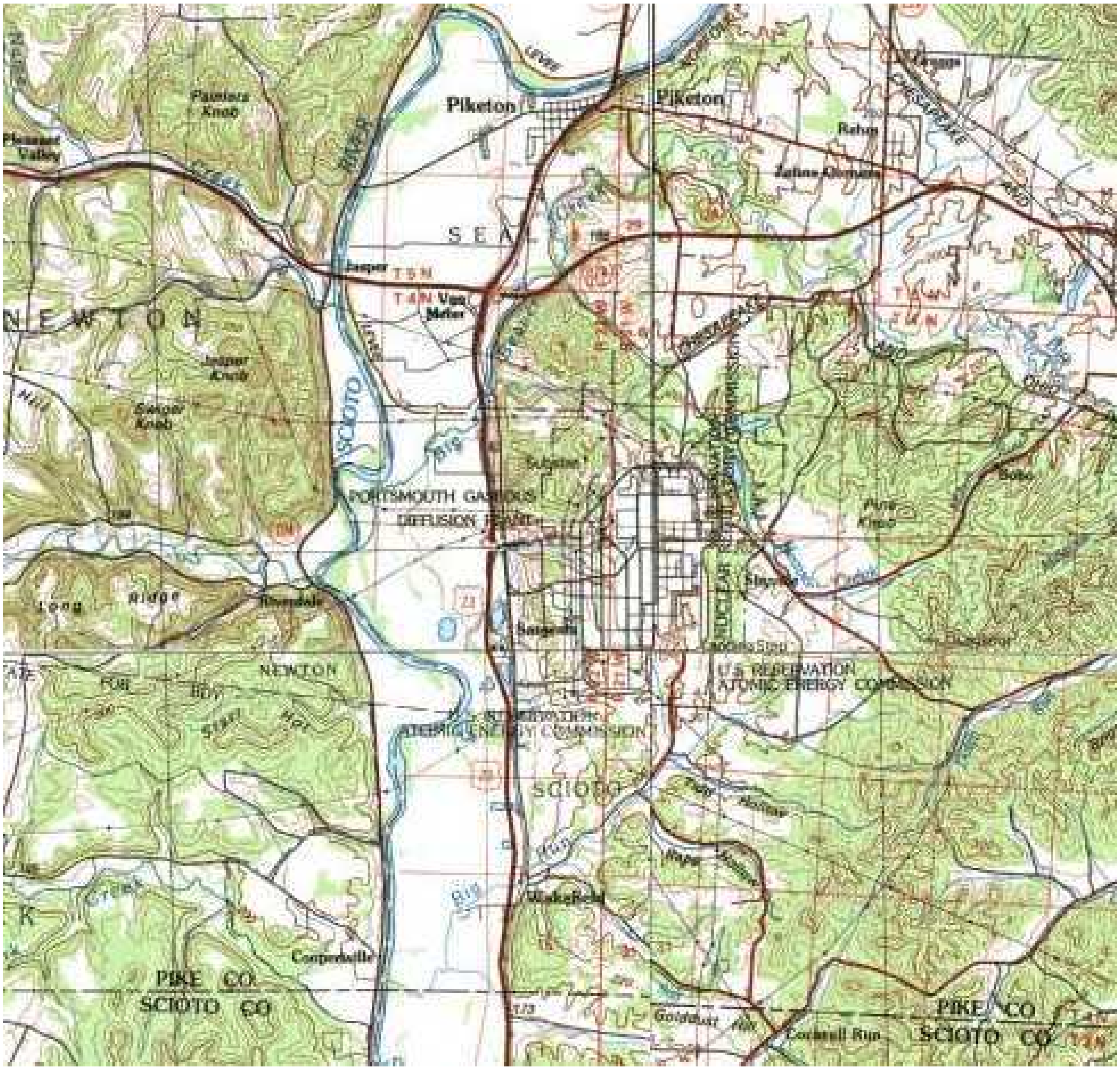

Figure 8.1-3
Topographical Map of PORTS
(Reference 1)

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 8.1-4
Aerial Map of the PORTS Reservation
(Reference 1)

8.2 References

1. U.S. Department of the Interior, U.S. Geological Survey, Reston, VA, Website: http://www.usgs.gov/index.html

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

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

4. USEC-02, Application for United States Nuclear Regulatory Commission Certification, Portsmouth Gaseous Diffusion Plant, Emergency Plan

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

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9.0 ENVIRONMENTAL PROTECTION

The Lead Cascade is located at the Portsmouth Gaseous Diffusion Plant (PORTS), within an existing nuclear facility with radioactive effluent control and as low as reasonably achievable (ALARA) programs that meet U.S. Nuclear Regulatory Commission (NRC) requirements. The Lead Cascade will not use any radionuclide not already used at PORTS and is expected to make a minimal contribution to site effluents. The Lead Cascade Environmental Protection Program, which is described below, is based on the PORTS program. The Lead Cascade program takes advantage of the well-established programmatic elements and experience. This approach will provide protection to the public and the environment while minimizing the chance of human or programmatic error. The Technical Services Manager is responsible for the Lead Cascade Environmental Protection Program, with support and resources provided by PORTS. PORTS provides resources and necessary programmatic support to the Environmental Protection Program as requested.

9.1 Environmental Report

The regulatory requirements for an Environmental Report are contained in 10 *Code of Federal Regulations* (CFR) Part 51. The NRC promulgated these regulations to implement the *National Environmental Policy Act* of 1969, which requires an assessment of the environmental impacts associated with all major Federal actions. For licensing actions that are not categorically excluded, the NRC conducts an independent assessment on the basis of the information submitted in the Environmental Report.

An Environmental Report meeting the requirements of 10 CFR 51.45 was prepared for the Lead Cascade and was submitted for review as a document titled, Environmental Report for the American Centrifuge Lead Cascade Facility. A supplement to the Environmental Report has been prepared in accordance with 10 CFR 51.60 and submitted for Lead Cascade license renewal.

9.2 Environmental Protection Measures

9.2.1 Radiation Protection Program

The Lead Cascade observes the following policies:

- The dose to members of the public resulting from gaseous emissions and liquid effluents shall be maintained in accordance with the ALARA principle and below regulatory limits.

- It is the responsibility of each employee to conduct his/her activities in such a manner so as to prevent or minimize the discharge of radioactive materials to the environment, and to report any unusual or excessive discharge of such material.

9.2.1.1 Radiological (ALARA) Goals for Effluent Control

The Lead Cascade, in conjunction with PORTS, maintains and uses gaseous and liquid effluent treatment systems, as appropriate, to maintain releases of radioactive material to unrestricted areas below the limits specified in 10 CFR 20.1301 and 40 CFR Part 190 and in accordance with the ALARA policy described below. Gaseous effluent control systems are also used to maintain releases of radioactive material to unrestricted areas below the dose constraint in 10 CFR 20.1101 and the dose limit in 40 CFR 61.92. Unrestricted areas are those areas beyond the U.S. Department of Energy (DOE) reservation boundary and to which any member of the public has unrestricted access.

The ALARA goal for airborne radioactive releases from PORTS is five percent of the NRC constraint level (10 CFR 20.1101) and Environmental Protection Agency (EPA) limit (40 CFR 61.92), or an annual Total Effective Dose Equivalent (TEDE) of 0.5 millirem (mrem) to the most exposed member of the public, as calculated under the PORTS Environmental Protection Program. This is also less than 15 percent of the most restrictive limit under 40 CFR Part 190, based on site experience. The EPA regulations explicitly require all airborne releases from the site to be included under the 40 CFR Part 61 public dose limit. This ALARA goal includes all PORTS airborne releases, including those associated with the Lead Cascade.

The ALARA goal for liquid radioactive releases from PORTS is 10 percent of the airborne ALARA goal, or an annual TEDE of 0.05 mrem to the most exposed member of the public, as calculated under the PORTS Environmental Protection Program. This is equivalent to 0.05 percent of the 10 CFR 20.1301 limit on annual public dose. All process liquid releases from the Lead Cascade will be collected and later pumped to existing treatment systems/discharge points on the PORTS reservation and will not be distinguishable from releases from other activities onsite. Storm water runoff from the Lead Cascade flows to two holding ponds. This ALARA goal includes all PORTS liquid releases, including those associated with the Lead Cascade.

9.2.1.2 Effluent Controls

9.2.1.2.1 Control of Airborne Effluents

All routine gaseous effluents from the Lead Cascade flow through the purge vacuum (PV) or evacuation vacuum (EV) systems as described in Section 1.1 of this license application. Both systems discharge to the X-3001 process vent through chemical absorbent (activated alumina trap media) traps. Four alumina traps are supplied to capture uranium hexafluoride (UF_6) in the effluent gas streams from the PV and EV systems. A set of four traps is designed to trap approximately 80 pounds of UF_6 and are changed out based upon operating experience.

Valving and piping allow the EV system throughput to bypass the chemical traps during the initial pump down of machines that have not been previously exposed to UF_6. Otherwise, the EV system throughput will pass through the chemical traps along with PV system throughput.

The process vent has gas flow monitoring instrumentation with local readout as well as the analytical instrumentation required to continuously monitor and to alarm UF_6 breakthrough in the effluent gas stream. The continuous vent sampler draws a flow proportional sample of the vent stream through two alumina traps in series by way of an isokinetic probe. Both vent and sampler flows are monitored by the samplers. The primary sample trap is also equipped with an automated radiation monitor to continuously monitor the accumulation of uranium in the sampler. This provides a real-time indicator of effluent levels for operational control of the gaseous effluent control systems.

The primary sample traps are replaced weekly and the secondary traps are replaced quarterly. The traps are returned to the PORTS laboratory for analysis of the alumina trap media. In the event of an unplanned or seriously elevated release, the applicable samplers are changed out (for immediate analysis) as soon as the situation has stabilized. Alternatively, the sampling period may be extended, provided the sampler is operating at all times the vent is operating.

Centrifuges may be disassembled for repair or inspection on the static stand in X-7726 (or in appropriate areas of the X-3001 or X-7725 if use of X-7726 equipment is not required). The extent to which a centrifuge is disassembled depends upon the nature of the fault. Assemblies removed in the repair process are tested and recertified before reinstallation. Centrifuges requiring repair or examination that have been in service will be opened using appropriate personnel protective equipment (PPE), and may also include engineered local ventilation systems to capture any residual uranium. Consequently, there should be no airborne radioactive effluents from X-7726 or any of the support buildings (e.g., X-7725, X-7727H, and X-3012). The workspace air in areas that may have airborne uranium is monitored as described in Section 4.7 of this License Application. Environmental personnel review summaries of the monitoring data quarterly to verify that ventilation exhausts are insignificant as defined in the Standard Review Plan (i.e., less that 3 x 10^{-13} µCi/mL uranium).

9.2.1.2.2 Control of Liquid Effluents

Sanitary wastewater from the Lead Cascade is treated at the X-6619 Sewage Treatment Plant (STP), which is located on the PORTS reservation and leased by USEC. The X-6619 was originally designed and constructed in the early 1980s to treat sanitary and process wastewater from both the existing gaseous diffusion plant (GDP) and the DOE Gas Centrifuge Enrichment Plant (GCEP) and has ample capacity to accept the Lead Cascade wastewater without either physical modification or adjustment to its discharge limits.

The X-6619 (National Pollutant Discharge Elimination System [NPDES] Outfall 003) uses screening and a grit chamber as preliminary treatment followed by an activated sludge treatment system. Mixed liquid from the aeration basins is clarified, filtered using multimedia sand filters, and then chlorinated/dechlorinated. Sludge is aerobically digested and dried on sludge drying beds. Sludge produced by the facility is drummed and stored pending future disposal, and the effluent discharges directly to the Scioto River.

The Lead Cascade centrifuges and PV and EV vacuum pumps are cooled by a closed-loop Machine Cooling Water (MCW) system to minimize the amount of water potentially

contaminated by uranium. There is no routine blowdown from the MCW system. Waste heat from the MCW system is discharged via heat exchanger to the GCEP Recirculating Cooling Water (RCW) system, which is cooled by a single cooling tower and discharges its blowdown water to the GDP RCW system. In the event that the Plant Cooling Tower is unavailable, sanitary water may be used for emergency cooling to the MCW system with the return water being discharged to the sanitary sewer system.

The GDP RCW System has ample capacity to accept the Lead Cascade effluents without either physical modification or adjustment to its discharge limits. Discharges from the GDP RCW System are monitored in accordance with the Environmental Protection Plan described in USEC-02. The GDP RCW blowdown also discharges directly to the Scioto River via a second underground pipeline. Details of the RCW systems can be found in Sections 5.1 and 3.4 of USEC-02.

Leakage from the MCW system, as well as incidental spills of water elsewhere in the Lead Cascade, is collected by the Liquid Effluent Collection (LEC) System. The LEC System consists of a set of drains and underground collection tanks for the collection and containment of leaks and spills of chemically treated water. The drains are located throughout the Lead Cascade. The tanks have a capacity of 550 gallons (Gal) each and are monitored by liquid level gauges mounted above grade on pipe stands. Water accumulated in the LEC tanks is sampled and pumped to either the X-6619 or containerized for disposal, depending on the results. Inventory monitoring of the tank contents will be used to detect any leaks.

Storm water runoff from the X-7725, X-7726 and northern portion of the X-7727H and the associated parking areas flows to the X-2230N West Holding Pond. No changes in these discharges are expected due to Lead Cascade installation. The X-2230N West Holding Pond provides a quiescent zone for settling suspended solids and dissipation of chlorine, and oil containment. The pond discharges to the Scioto River via an unnamed creek south of the Main Access Road. A composite sampler is utilized on the pond’s discharge, comparable to the composite samplers used on the other leased outfalls.

Storm water runoff from the X-3001, X-3012 and southern portion of the X-7727H and the associated parking areas flows to the X-2230M Southwest Holding Pond. No changes in these discharges are expected due to Lead Cascade installation. The X-2230M Southwest Holding Pond provides a quiescent zone for settling suspended solids and dissipation of chlorine, and oil containment. The pond discharges to the Scioto River via an unnamed creek south of the Southwest Access Road. A composite sampler is utilized on the pond’s discharge comparable to the composite samplers used on the other leased outfalls.

9.2.1.3 ALARA Reviews and Reports to Management

Action levels for control of both gaseous and liquid radioactive effluents from PORTS have been established based on the ALARA philosophy. The action levels described in Table 9.2-1 ensure operational control system deficiencies are documented and acted upon in a responsible manner and in a timeframe to remain well within the regulatory limits and below ALARA goals. The Baseline Effluent Quantity (BEQ) used in Table 9.2-1 is the maximum effluent expected under normal operating conditions. BEQs have been established by the

responsible facility manager and the responsible PORTS personnel, for every continuously monitored radiological vent and liquid discharge point to unrestricted areas. These BEQs are reviewed annually, at a minimum, by the responsible facility manager, ALARA Committee, and responsible PORTS personnel, to ensure the principles described in the facility's ALARA policy are followed. Recommendations for changes are reviewed by Lead Cascade senior management. The Operations Shift Supervisor and the Plant Shift Superintendent are jointly responsible for assuring that action levels are acted upon.

The BEQs established for the new release points associated with the Lead Cascade are listed in Table 9.2-2. BEQs for GDP release points are not expected to change due to Lead Cascade operations. The uranium BEQ for the Lead Cascade process vent is based on engineering estimates of the maximum weekly activity of uranium that will be released by the facility. Because the Lead Cascade does not expect to discharge any licensed material through its holding ponds, the BEQs for the two ponds are on a concentration basis. The concentration-based BEQs were selected to be less than ten percent of the applicable discharge limits in 10 CFR Part 20 Appendix B and at least five times the minimum detectable concentration/activity.

9.2.1.4 Waste Minimization

Lead Cascade waste minimization and pollution prevention activities are coordinated by the Technical Services Manager, with support from PORTS.

It is the policy of the Lead Cascade to:

- Promote the use of non-hazardous materials in plant operations to minimize the potential risk to human health and the environment.
- Reduce or eliminate pollution to all media (i.e., land, water, and air) at the source to the lowest reasonably achievable level through material substitution, process optimization and innovation, in-process closed-loop recycle, and waste segregation.
- Use, reuse, reclaim, or recycle to the maximum extent practical those waste streams that cannot be eliminated or minimized by source reduction.
- Promote the continual evaluation and implementation of waste minimization and pollution prevention opportunities in ongoing plant operations, technical support activities, and project design.
- Develop in all employees an awareness of environmental problems through participation in the Waste Minimization and Pollution Awareness Program activities.
- Conduct and implement waste minimization activities with full regard to requirements for quality, productivity, safety, and environmental compliance.

To this end, activities will be evaluated for waste minimization opportunities with emphasis on those that generate hazardous wastes, hazardous and radioactive wastes, and low-

level radioactive wastes (LLRW). Waste that is nevertheless generated will be treated to the extent practical to reduce the volume, toxicity, or mobility before storage or disposal.

This applies to facility operations, associated support operations, and site subcontractors that generate waste.

9.2.2 Effluent and Environmental Monitoring

9.2.2.1 Airborne Effluents

9.2.2.1.1 Anticipated Effluent Levels

The maximum anticipated gaseous effluents from the Lead Cascade have been modeled using the EPA-approved and distributed dispersion/dose assessment model, CAP88-PC, and onsite meteorological data. The maximum gaseous effluent anticipated under normal operations is 0.1 millicuries (mCi) of uranium over a week, or up to 5.2×10^{-3} curie (Ci) per year. Using the 2001 meteorological data from the original Lead Cascade license application; the projected maximum airborne concentration of total uranium due to Lead Cascade operations is 2.1×10^{-16} microcuries per milliliter (µCi/mL), with an associated TEDE of 0.023 mrem. Using the most recent meteorological data (calendar year 2009); the projected maximum airborne concentration of total uranium due to Lead Cascade operations fell to 9.7×10^{-17} µCi/mL, with an associated TEDE of 0.011 mrem. Both of these uranium concentrations are at least four orders of magnitude lower than the airborne uranium values in Table 2 of 10 CFR Part 20 Appendix B (3×10^{-12} µCi/mL) and far below the ALARA goal given in Section 9.2.1.1 above. These results are summarized in Table 9.2-3.

Actual uranium releases from the Lead Cascade over the last three years have been substantially lower than the postulated maximum, ranging from 7.6×10^{-6} to 8.8×10^{-6} Ci per year with associated public TEDEs ranging from 0.0032 to 0.0058 microrem. In all cases, the maximum exposed individual (MEI) for the Lead Cascade itself is located in the south-southwest sector of the reservation boundary.

Due to other operations at PORTS and site geometry, the site MEI is located in the east-northeast sector of the site boundary. The prevailing downwind direction at PORTS is northeasterly, but the Lead Cascade's location in the far southwestern corner of the DOE reservation results in a much greater dispersion distance and lower exposures to the public in that direction. The EPA regulation that drives the annual dose assessment described in Section 9.3.1 applies to the entire site, so only the site MEI is discussed in the annual dose assessment. In addition, EPA regulations and policies require USEC to account for technetium emissions from all vents, even though the uranium used in the Lead Cascade was specifically selected to be free of technetium contamination. Undetectable technetium concentrations are presumed to be equal to the Minimum Detectable Activity (MDA), which are three orders of magnitude higher than the uranium concentrations in the Lead Cascade vent. The public doses cited above have been adjusted to remove the effect of these presumed technetium emissions.

9.2.2.1.2 Demonstration of Compliance

Characterization of the radiological consequences of radionuclides released to the atmosphere from PORTS is accomplished by annually calculating the TEDEs to the maximally exposed person and to the entire population residing within 80 kilometers (km) (50 miles) of the plant. This approach is mandatory under the EPA regulations at 40 CFR Part 61 and has been accepted by the NRC for the existing operations at PORTS.

Annual radionuclide releases to air are measured by the continuous vent samplers described in Section 9.2.2.1.3 of this license application or estimated in accordance with guidance in 40 CFR Part 61 Appendices D and E. Atmospheric dispersion of the releases is modeled and the consequent public radiation dose is estimated using the EPA approved computer models in accordance with EPA guidance. An annual report summarizing the atmospheric releases and the dose assessment results is submitted in accordance with 40 CFR Part 61 Subpart H and EPA guidance. In accordance with EPA requirements, these calculations include all gaseous radioactive effluents at PORTS.

The dose calculations are made using the CAP88 package of computer codes. This package contains an approved version of the AIRDOS-EPA and DARTAB computer codes and the ALLRAD88 radionuclide data file. The latest version (CAP88-PC Version 3) has updated the original ALLRAD88 data with the data from Federal Guidance Report No. 11 (FGR 11). The AIRDOS-EPA computer code implements a steady-state, Gaussian plume, atmospheric dispersion model to calculate concentrations of radionuclides in the air and on the ground based on radionuclide releases to the atmosphere and annualized meteorological data. It then uses 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 I* (October 1977), food-chain models to calculate radionuclide concentrations in foodstuffs (e.g., vegetables, meat, milk) and subsequent intakes by individuals. The DARTAB computer code then uses these calculated uptakes and radionuclide data from the ALLRAD88 data file to calculate annual radiation doses to members of the public.

The annualized meteorological data used in the calculations consist of joint frequency stability array distributions of wind direction, wind speed, and atmospheric stability that is prepared from data collected from the site meteorological tower. Data from the National Weather Service may be used in lieu of or to supplement onsite meteorological data. PORTS has a consistent annual pattern of low-level southwesterly winds predominating over the year. During the winter season, northeasterly winds are common though. This is largely attributable to the channeling effect of the hills and ridges on either side of the reservation, which runs roughly southwest to northeast.

EPA published default values for offsite parameters (such as local crop productivity) are used in the AIRDOS-EPA model and, in accordance with EPA recommendations, rural patterns for food sources (i.e., home grown versus local production versus national supermarket chains) are assumed.

9.2.2.1.3 Monitoring of Gaseous Release Points

Quantifiable gaseous radioactive effluents from the Lead Cascade are limited to the X-3001 north process vent, which serves both the EV and PV systems. The gaseous effluent monitoring system for the Lead Cascade consists of a continuous vent sampler similar to those used on the GDP radionuclide vents. Vent samples are collected and managed in the same manner as the GDP vent samples.

The vent samplers draw a flow proportional sample of the vent stream through two alumina traps in series by way of an isokinetic probe. Both vent and sampler flows are monitored by the samplers. The primary sample traps are replaced weekly and the secondary traps are replaced quarterly. In the event of an unplanned or seriously elevated release, the applicable samplers are changed out (for immediate analysis) as soon as the situation has stabilized. Alternatively, the sampling period may be extended, provided the sampler is operating at all times the vent is operating. The primary sample trap is also equipped with an automated radiation monitor to continuously monitor the accumulation of uranium in the sampler. This provides a real-time indicator of effluent levels for operational control of the gaseous effluent control systems.

Vent samples are analyzed for ^{234}U, ^{235}U, ^{238}U and ^{99}Tc concentrations. A total uranium activity and technetium activity is calculated from the isotopic concentrations and published specific activities. No technetium is expected to be present in the Lead Cascade process vent, since the Licensee does not intend to introduce any technetium to this facility. Technetium is known to be a potential contaminant in the GDP vents however, so all vent samples are analyzed for it. Specific details of the analytical methods are presented in Section 9.2.2.6 of this license application.

Centrifuges may be disassembled for repair or inspection on the static stand in X-7726 (or in appropriate areas of the X-3001 if use of X-7726 equipment is not required), which does not have a process vent. Centrifuges requiring repair or examination that have been in service will be opened using appropriate PPE, and may also include engineered local ventilation systems to capture any residual uranium. Consequently, there should be no airborne radioactive effluents from X-7726 or any of the support buildings (e.g., X-7725, X-7727H, and X-3012) to measure. The workspace air in areas that may have airborne uranium is monitored as described in Section 4.7 of this License Application. Environmental personnel review summaries of the monitoring data quarterly to verify that ventilation exhausts are insignificant as defined in the Standard Review Plan (i.e., less than 3×10^{-13} μCi/mL uranium).

9.2.2.1.4 Action Levels

Action levels for control of gaseous radioactive effluents from operations have been established based on the ALARA philosophy. The action levels described in Table 9.2-1 ensure operational control system deficiencies are documented and acted upon in a responsible manner and in a timeframe to remain well within the regulatory limits and below ALARA goals. The BEQs used in Table 9.2-1 is the maximum effluent expected under normal operating conditions. BEQs have been established for every continuously monitored radiological vent. The specific BEQs established for the Lead Cascade process vent are listed in Table 9.2-2.

9.2.2.1.5 Other Permits and Licenses

All new air pollutant sources or modifications of existing sources in the State of Ohio are required to have a Permit to Install (PTI) from the Ohio EPA prior to installation of the source. The Licensee obtained one PTI for the Lead Cascade process vent as well as PTIs for twenty ancillary units such as feed carts and HEPA-filtered vacuum cleaners. (The ancillary units would be exempted from PTIs as "trivial" sources if they did not have the potential for some trace of radioactive material in their exhausts.) Within one year of the PTI for the Lead Cascade process vent being issued, the Licensee applied to the Ohio EPA for a modification to its Title V permit to incorporate the entire Lead Cascade into the existing permit. The Title V permit supersedes the PTI once it is modified. As stated in the Supplement to the Environmental Report, Section 1.0, all these PTIs have been issued by the OEPA and the latest update application for Title V Permit was submitted on June 8, 2010.

All sources of airborne radionuclides at DOE-owned facilities are covered by an EPA Permit-By-Rule issued under 40 CFR Part 61 (National Emission Standards for Hazardous Air Pollutants [NESHAP]) Subpart H. This rule imposes a limit on airborne effluents of 10 mrem/year to the MEI, which applies to the entire site regardless of who "owns" any individual source within the site. The rule also requires an annual report, submitted by June 30 of each year, detailing the processes at the site, the airborne effluents from each source, and annual TEDE to the MEI as calculated by a method approved by the EPA. Since the United States Enrichment Corporation (Corporation) and DOE operations on the PORTS reservation are largely independent of each other, both organizations submit separate annual reports and include each other's calculated TEDE along with their own. EPA Region V has accepted this as an adequate demonstration of compliance. A copy of the Corporation's report is provided to NRC in accordance with commitments for the GDP, as described in Section 9.3.1 of this license application.

Also under the NESHAP rule, all new or modified sources of airborne radionuclides at DOE-owned facilities are required to have prior Permission to Construct from EPA unless the change has a projected maximum public TEDE of less than 0.1 mrem/year. The Lead Cascade is below this threshold. Consequently, the Licensee only needed to give the EPA a timely notification of its intent to install and operate the Lead Cascade to comply with this requirement.

9.2.2.2 Liquid Effluents

9.2.2.2.1 Anticipated Effluent Levels

Anticipated routine liquid radioactive effluents from the Lead Cascade are minimal. The bulk of liquid radioactive effluents are decontamination and cleaning solutions. Centrifuges will not be routinely changed out, but routine maintenance such as instrument repair or repair to the PV and EV Evacuation Systems occur. There are also maintenance activities that could require cleaning and/or decontamination.

Sanitary wastewater and leakage or spillage from the Lead Cascade is treated at the existing X-6619 (Outfall 003 on USEC's NPDES Permit) as described in Section 9.2.1.2.2 of this license application, then discharges directly to the Scioto River in accordance with 10 CFR 20.1301 and the Environmental Protection Plan described in USEC-02. Lead Cascade effluents are not expected to make a significant difference in the existing release levels.

There are no anticipated effluents from the MCW system, because it is designed as a closed-loop system. There will be an increase in the heat load to the GCEP RCW System but this should not change the blowdown rate significantly. Blowdown from the GCEP RCW System is processed through the existing GDP RCW system (Outfall 004 on USEC's NPDES Permit) as described in Section 9.2.1.2.2 of this license application, then discharges directly to the Scioto River in accordance with 10 CFR 20.1301 and the Environmental Protection Plan described in USEC-02.

Storm water runoff flows through the X-2230N and X-2230M holding ponds (Outfalls 012 and 013) as described in Section 9.2.1.2.2 of this license application, then discharges to the Scioto River via separate unnamed creeks in accordance with 10 CFR 20.1301. Lead Cascade effluents are not expected to make a significant difference in the existing release levels.

Anticipated liquid radioactive releases from these four points are summarized in Table 9.2-4, along with the limits from Table 2 of 10 CFR Part 20 Appendix B for comparison. Note that these discharges include all site contributions. All of the anticipated discharge levels are at least one order of magnitude below the Appendix B, Table 2 limits even before they mix with the Scioto River. Activity concentrations in the table are based on weekly composite samples from January 2007 through December 2009. The activity releases are based on the BEQs, which are the maximum anticipated weekly discharge. Historically, annual average discharges are well below the BEQs. Because the Lead Cascade does not expect to discharge any licensed material through its holding ponds, the BEQs for the two ponds are on a concentration basis. The concentration-based BEQs were selected to be less than ten percent of the applicable discharge limits in 10 CFR Part 20 Appendix B and at least five times the minimum detectable concentration/activity.

9.2.2.2.2 Demonstration of Compliance

Characterization of the consequences of radionuclides released to surface water from PORTS is accomplished by annually calculating the TEDE to the maximally exposed person using 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 I* (October 1977).

Water is sampled at leased site water outfalls and at selected surface water locations, including upstream and downstream locations in the Scioto River. A hypothetical dose is calculated using measured radionuclide discharges from outfalls and the average annual flow of the Scioto River. Conservative public utilization estimates for the Scioto River are used in the calculations.

Waterborne radionuclide releases from leased outfalls flow to the Scioto River, and treated process wastewater flows directly to the river via buried pipelines. Consequently, the Licensee considers the Scioto River downstream of plant discharges to be the point of maximum public exposure to radionuclides released from the plant.

Regulatory Guide 1.109 food chain models are used to calculate radionuclide intakes by individuals. The pathways considered are the direct use of surface water for drinking water, the ingestion of fish from contaminated water, and the use of contaminated water for irrigation. Radionuclides released to any receiving stream are presumed to be evenly dispersed throughout the river (e.g., no depletion by settling or absorption, etc.).

Irrigation is not a widespread practice in the lower Scioto Valley, and when it is practiced, groundwater is the preferred source. Furthermore, analyses of local crops have never shown any detectable radionuclide contamination. Consequently, the contribution to the public radiation dose via the irrigation pathway is zero.

There are no public or known private drinking water intakes on the Scioto River or on Big Beaver or Little Beaver Creek. However, in view of the rural and loosely regulated character of the lower Scioto Valley, the Licensee cannot discount the possibility that there is a small private drinking water intake somewhere along the lower Scioto River. Sport fishing is also common along the Scioto River, and while there is no data available on how much fish is consumed per person from this source, it could conceivably approach the 21 kilograms (kg) per year value cited for the International Commission on Radiological Protection's reference adult. Consequently, PORTS calculates an annual TEDE for a hypothetical person drinking 730 liters (L) per year of untreated river water and eating 21 kg of fish per year caught in the Scioto River.

There is no routine comparison between measured waterborne concentrations in the Scioto River and those predicted by this model because there is no statistically significant difference between upstream and downstream measured concentrations.

Radioactive materials released to the site sewers never leave the control of the Licensee until they are released from the STP under 10 CFR 20.1301.

9.2.2.2.3 Monitoring of Liquid Release Points

There are eight leased water outfalls to offsite surface waters relating to the operation of the GDPs as well as X-2230N and X-2230M for the Lead Cascade. These outfalls are monitored to quantify radiological waterborne effluents. The locations of the outfalls are shown in Figure 9.2-1 (located in Appendix B of this license application). Only the four outfalls discussed above (Outfalls 003, 004, 012 and 013) are potentially impacted by Lead Cascade effluents.

Radiological analyses are performed on samples from leased outfalls. Outfalls with routine continuous flow are monitored with composite samplers and analyzed weekly. Outfalls with intermittent flow are monitored with grab samples during periods of flow. Aliquots from these samples are analyzed for total uranium concentrations, gross alpha, gross beta, and technetium beta activities. The ratio of alpha activity to total uranium, along with process data, is used to calculate the proportions of the individual uranium isotopes. Specific details of the analytical methods are presented in Section 9.2.2.5 of this license application.

The only underground tanks in the Lead Cascade are the tanks of the LEC system. The LEC system consists of drains and collection tanks used primarily for collecting leaks and spills of chemically treated water. The drains are located throughout the Process Building. The tanks have a capacity of 550 Gal each. The tanks are monitored by liquid level gauges mounted above grade on pipe stands. Routine monitoring of the tanks is based on observing and tracking the levels indicated on the gauges.

9.2.2.2.4 Action Levels

Action levels for control of liquid radioactive effluents from PORTS have been established based on the ALARA philosophy. The action levels described in Table 9.2-1 ensure operational control system deficiencies are documented and acted upon in a responsible manner and in a timeframe to remain well within the regulatory limits and below ALARA goals. The BEQs used in Table 9.2-1 are the maximum effluents expected under normal operating conditions. BEQs have been established for every liquid discharge point to unrestricted areas. The BEQs established for the GDP-associated liquid release points are listed in Table 5.1-4 of USEC-02. The BEQs established for X-2230M and X-2230 N are listed in Table 9.2-2.

9.2.2.2.5 Other Permits and Licenses

All point discharges to waters of the State of Ohio are required to be authorized under a NPDES Permit issued by the Ohio EPA. There are three NPDES Permits issued to the PORTS site, one to DOE, one to Uranium Disposition Services (UDS) and one to USEC for their respective discharges.

A special condition of the Corporation's NPDES Permit is to provide a quarterly report to the Ohio EPA summarizing the radioactive discharges from the outfalls. A copy of this report is available to the NRC as described in Section 9.3.2 of this license application.

9.2.2.3 Waste Management Systems

The Lead Cascade is located within an existing nuclear facility with a Radioactive and Mixed Waste Program in place. Waste generated by Lead Cascade construction and operation is managed with support from PORTS.

9.2.2.3.1 Waste Segregation and Collection

Licensee-generated wastes are collected and packaged, where feasible, by the waste generator. Wastes known to be suitable for release to unrestricted areas based on the point and process of generation are segregated at the source, when possible, from wastes not suitable for release to unrestricted areas. Wastes from areas controlled for loose radioactive contamination are considered to be potentially contaminated until characterized. Wastes requiring characterization to determine whether they may be released to unrestricted areas are segregated upon completion of such characterization.

9.2.2.3.2 Waste Packaging and Labeling

Containers known to contain radioactive waste, including packaging, are labeled in accordance with procedural requirements.

Waste is packaged in appropriate containers to meet U. S. Department of Transportation (DOT) and 10 CFR Part 71 requirements. Some general types of waste packaging include, but are not limited to:

- Solid Waste 5-, 30-, 55-, or 110-Gal drums
- Liquid Wastes 5-, 30-, or 55-Gal drums or 330-Gal totes
- Corrosives, Acids Polybottles or polydrums
- Scrap Metal B-25 boxes or other similar boxes; various drums

In addition, 85- and 110-Gal overpacks may be used for appropriate wastes and damaged containers.

9.2.2.3.3 Radioactive Waste Storage

Those Lead Cascade wastes that are regulated for radiological content only are removed from the generating facility and stored at a radioactive waste storage facility prior to final disposal. PORTS-generated mixed wastes are stored locally for up to 90 days pending treatment, disposal, or transfer. The State of Ohio has adopted a federal exemption to the hazardous waste rules that is available under the 40 CFR Part 266, Subpart N. This exemption delegates the Federal EPA's authority over mixed waste to the NRC, subject to certain conditions. As of this writing, the Licensee has not needed to use this exemption. Other areas may be utilized as waste storage facilities as required by facility operations. If outdoor storage is necessary, radioactive wastes with removable contamination are packaged in containers, wrapped or covered to prevent the release of radioactivity. Storage areas are posted in accordance with procedural requirements.

Access to waste storage containers is restricted to trained personnel in accordance with 10 CFR 20.1905. Containers are inspected quarterly, at a minimum, to ensure container integrity and to identify and correct any leaks or other problems.

9.2.2.3.4 Radioactive Waste Treatment

Mixed aqueous wastes that cannot be processed in the Licensee facilities are stored onsite until treatment is available at commercial treatment facilities that are licensed in accordance with 10 CFR Part 61, or applicable NRC Agreement State requirements.

9.2.2.3.5 Offsite Waste Shipments

Offsite shipments of radioactive wastes are manifested in accordance with 10 CFR 20.2006. Waste shipments are packaged, labeled, and manifested in accordance with applicable State, DOT, NRC, and EPA requirements.

9.2.2.3.6 Waste Disposal

Licensee-generated wastes are disposed of at commercial disposal facilities that are licensed in accordance with 10 CFR Part 61 or applicable NRC Agreement State requirements. Packages are inspected prior to shipment, as appropriate, to verify compliance with applicable packaging and transportation requirements. Copies of the disposal site license are retained in accordance with procedural requirements.

Waste disposals are in compliance with 10 CFR Part 20, Subpart K. Waste disposal records are retained in accordance with 10 CFR 20.2108. Classified waste is disposed of in accordance with 10 CFR Part 95 and security program requirements.

9.2.2.3.7 Waste Tracking and Documentation

LLRW and mixed waste generated at PORTS is tracked through a Request for Disposal system. Each waste container is given a unique identification number. The identification numbers are entered and maintained in a computer-based database. The database is updated to reflect location, characterization, and waste disposal information.

9.2.2.3.8 Other Permits and Licenses

Presently, PORTS is classified as a large-volume generator of *Resource Conservation and Recovery Act* of 1976 (RCRA) hazardous and mixed (i.e., RCRA hazardous and LLRW) wastes which transfers wastes to appropriately permitted Treatment, Storage, and Disposal (TSD) Facilities within 90 days, including the onsite DOE TSD Facility, as described in the Director's Final Findings and Orders. For the Lead Cascade, the Licensee has not utilized the federal exemption adopted by the State of Ohio (see Section 9.2.2.3) for mixed waste. No mixed waste has been generated or is anticipated to be generated during Lead Cascade operations.

9.2.2.4 Environmental Monitoring

The Lead Cascade is located within an existing nuclear facility with an Environmental Monitoring Program in place. As discussed above, the Lead Cascade uses only radionuclides already in use at PORTS and has no direct discharges to unrestricted areas except for its process vents. Consequently, no additional environmental monitoring beyond the scope of the existing environmental monitoring program and effluent monitoring of the Lead Cascade vents is conducted.

9.2.2.4.1 Air Monitoring

Between 1980 and 2007, annual gaseous uranium effluents from PORTS have ranged from 0.97 Ci to 0.001 Ci. Ambient air samples collected over this period by the Corporation, its predecessor organizations and DOE show that these levels of effluents do not produce a quantifiable difference in ambient air concentrations in unrestricted areas. Since the Lead Cascade contains less than 0.15 Ci of uranium and expects to release no more than 0.001 curie per week (Ci/wk), ambient air monitoring will not be useful in detecting a public impact due to gaseous effluents from the Lead Cascade. Therefore, atmospheric impacts of Lead Cascade operation, including action levels, will be estimated based on gaseous effluent monitoring and atmospheric dispersion modeling as described in Section 9.2.2.1 of this license application.

USEC ceased sampling ambient air and returned the site's network of permanent air samplers to DOE in 1999. DOE has upgraded the samplers and now uses them to monitor for atmospheric impacts from non-point releases from its environmental remediation projects. The most recent ambient air data publicly available is through 2007.

The site maintains a meteorological tower that is located on the southern section of the site. The tower is equipped with instruments at the ground, 10-, 30-, and 60-meter levels. Among the parameters measured are air temperature, wind speed, wind direction, relative humidity, solar radiation, barometric pressure, precipitation, and soil temperature. Data from the National Weather Service or other local sources may be used in lieu of or to supplement onsite data.

The effluent monitoring and meteorological data are used to calculate the potential environmental impacts of airborne effluents from PORTS facilities (including the Lead Cascade) using EPA-approved dispersion models as described in Section 9.2.2.1 of this license application.

9.2.2.4.2 Soil and Vegetation

Between 1980 and 2009, annual gaseous uranium effluents from PORTS have ranged from 0.97 Ci to 0.001 Ci. Soil and vegetation samples collected over this period by the Corporation and its predecessor organizations showed that these levels of effluents do not produce a statistically significant difference in soil and vegetation concentrations in unrestricted areas. (Liquid effluents have no measurable impact on soil and terrestrial vegetation around PORTS.) Since the Lead Cascade contains less than 0.15 Ci of uranium and expects to release no more than 0.001 Ci/wk, soil and vegetation monitoring will not be useful in detecting a public impact due to gaseous effluents from the Lead Cascade.

Soil and vegetation monitoring is more likely to be useful in assessing the long-term impacts of routine effluents from GDP operations or DOE environmental remediation projects; or in assessing the impact of an unplanned release from the GDP. The Corporation has committed to maintain a soil and vegetation monitoring program for these purposes.

Soil and vegetation (wide-blade grass, typical of local cattle forage) samples are collected semiannually to verify that airborne emissions from the site are not influencing the soil or vegetation surrounding the site. The sampling networks completely surround the facility, including the predominant downwind directions. Soil samples are analyzed for gross alpha activity, gross beta activity, technetium beta activity, and total uranium concentration. Vegetation samples are analyzed for technetium beta activity and total uranium concentration. Specific details of the analytical methods are presented in Section 9.2.2.5 of this license application. See Table 9.2-5 for a summary of the soil and vegetation results for the three years prior to Lead Cascade licensing (1999-2001) and the last three years of Lead Cascade operations (2007-2009).

In addition to the semiannual vegetation samples, the Corporation collects annual crop samples from local gardeners and farmers on a voluntary basis. Because of the voluntary nature of these samples, the sampling locations change from year to year. Crop samples are normally analyzed for technetium beta activity and total uranium concentration only. The analytical methods are the same as for the vegetation samples. No contamination has been found in crop samples.

9.2.2.4.3 Surface Water

Between 1980 and 2009, annual waterborne uranium effluents from PORTS have ranged from 0.71 Ci to 0.014 Ci. Surface water samples collected over this period by the Corporation and its predecessor organizations showed that these levels of effluents do not produce a statistically significant difference in the Scioto River. (All liquid effluents from the Lead Cascade flow to the Scioto River without passing through any other named waterways.) Because the Lead Cascade contains less than 0.15 Ci of uranium and does not expect any direct liquid effluents (except stormwater runoff) to unrestricted areas, ambient surface water monitoring will not be useful in detecting a public impact due to liquid effluents from the Lead Cascade. Therefore, all environmental impacts of Lead Cascade operation, including action levels, will be estimated based on effluent monitoring and pathways modeling as described in Section 9.2.2.2 of this license application.

Radiological analyses are performed on grab samples from upstream and downstream locations in Little Beaver Creek, Big Beaver Creek, Big Run Creek, and the Scioto River. Samples are collected weekly from the Scioto River and one downstream location in Little Beaver Creek. All other locations are sampled monthly. Of these streams, only the Scioto River is potentially impacted by liquid effluents from the Lead Cascade. The river is sampled weekly and analyzed for the same parameters as the site outfalls: gross alpha, gross beta, and ^{99}Tc beta activities and total uranium concentrations. Specific details of the analytical methods are presented in Section 9.2.2.5 of this license application. See Table 9.2-6 for a summary of the surface water results for the three calendar years prior to Lead Cascade licensing (1999-2001) and the last three years of Lead Cascade operations (2006-2009).

9.2.2.4.4 Sediment Monitoring

Between 1980 and 2009, annual waterborne uranium effluents from PORTS have ranged from 0.71 Ci to 0.014 Ci. Sediment samples collected over this period by the Corporation and its predecessor organizations showed that these levels of effluents do not produce a statistically significant difference in the Scioto River. (All liquid effluents from the Lead Cascade flow to the Scioto River without passing through any other named waterways.) Because the Lead Cascade contains less than 0.15 Ci of uranium and does not expect any direct liquid effluents (except stormwater runoff) to unrestricted areas, sediment monitoring will not be useful in detecting a public impact due to liquid effluents from the Lead Cascade. Therefore, all environmental impacts of Lead Cascade operation, including action levels, will be estimated based on effluent monitoring and pathways modeling as described in Section 9.2.2.2 of this license application.

Sediment monitoring is more likely to be useful in assessing the long-term impacts of routine effluents from GDP operations or DOE environmental remediation projects; or in assessing the impact of an unplanned release from the GDP. The Corporation has committed to maintain a sediment monitoring program for these purposes.

Sediment sampling around the site is conducted semiannually to assess potential radionuclide accumulation in the surrounding receiving streams. Locations include the outfalls from the X-2230M and X-2230N. Sediment sample analyses include gross alpha activity, gross beta activity, and technetium beta activity and total uranium concentration. Specific details of the analytical methods are presented in Section 9.2.2.5 of this license application. See Table 9.2-6 for a summary of the sediment results over the three years prior to Lead Cascade licensing (1999-2001) and the last three years of Lead Cascade operations (2006-2009).

9.2.2.4.5 Groundwater

Due to historical operations, the PORTS site has multiple plumes of groundwater contamination. The primary contaminant in all of the plumes is the halogenated solvent trichloroethylene, but limited areas of technetium contamination also exist. Under the United States Enrichment Corporation Privatization Act, DOE is responsible for all pre-existing conditions.

DOE is conducting a site-wide environmental remediation program under an Agreed Order with the State of Ohio. As part of this program, all site groundwater monitoring is under the control of DOE. Consequently, the Licensee does not propose to conduct additional groundwater monitoring at PORTS as part of Lead Cascade operations.

9.2.2.4.6 Direct Gamma Radiation Monitoring

The Lead Cascade will not introduce any additional sources of gamma radiation to the site, consequently no change in direct radiation levels will occur as a result of the project. Direct radiation from existing sources is monitored according to the Environmental Monitoring Plan in USEC-02.

The site conducts an external gamma radiation monitoring program consisting of lithium fluoride thermoluminescence dosimeters (TLDs) positioned at various site locations and at locations offsite. There are nine dosimeters spaced around the perimeter of the limited area of the reservation; eight dosimeters spaced around the reservation boundary; and two dosimeters located offsite. These dosimeters are collected and analyzed quarterly. Processing and evaluation are performed by a processor holding current accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP) of the National Institute of Standards and Technology (NIST). See Table 9.2-7 for a summary of the TLD results for the three years prior to Lead Cascade licensing (1999-2001) and the last three years of Lead Cascade operations.

The only significant sources of gamma radiation at the site are the uranium isotope ^{235}U and the short-lived ^{238}U daughters. The Lead Cascade operates with a limited amount of UF_6 that is already onsite. Consequently, no new source of gamma radiation has been introduced to the site.

9.2.2.5 Laboratory Standards

Effluent and environmental analyses are carried out in the X-710 Analytical Laboratory, except for processing TLDs. TLDs are processed by an NVLAP-certified outside vendor. Priority is given to the vent samples because airborne effluents traditionally account for more than 99 percent of the annual public radiation dose at PORTS. The following discussion presents the analytical methodologies currently used to characterize effluent and environmental samples. Other methodologies may be utilized with equivalent or better sensitivity if and when they become available.

Vent samples (i.e., activated alumina trap media) are analyzed for uranium isotopes (^{234}U, ^{235}U, and ^{238}U) and ^{99}Tc. Uranium isotope concentrations are determined using either alpha spectrometry or Inductively Coupled Plasma/Mass Spectrometry (ICP/MS). Technetium concentrations are determined using liquid scintillation counting. Analytical results are reported in micrograms of analyte per gram of alumina. These results are converted to grams released by laboratory personnel using recorded flow data and the measured weight of alumina in the sampler and to activity using published specific activities. The Corporation routinely measures gaseous effluents equivalent to an annual public dose of well under 0.1 mrem. Because the airborne concentrations in 10 CFR Part 20 Appendix B, Table 2 are equivalent to an annual dose of 50 mrem, the Minimum Detectable Activity (MDA) of these methods are equivalent to less than 0.2 percent of the Appendix B, Table 2 values.

Water samples from NPDES outfalls are analyzed for gross alpha and gross beta activity, technetium beta activity, and total uranium concentration. The gross activities are determined by proportional counter and the technetium activity by liquid scintillation. The MDAs are 5×10^{-9} μCi/mL for gross alpha, 1.5×10^{-8} μCi/mL for gross beta, 2×10^{-8} μCi/mL for technetium beta. The total uranium concentration is determined by ICP/MS, with a minimum detectable concentration of 0.001 micrograms per milliliter (μg/mL). The isotopic distribution of the total uranium is estimated to match the calculated uranium alpha activity to the measured gross alpha activity. The Table 2 values for liquid releases are 3×10^{-7} μCi/mL for each of the uranium

isotopes and 6 x 10^{-5} μCi/mL for technetium. Consequently, all the MDAs for liquid effluents are less than two percent of the applicable 10 CFR Part 20 Appendix B, Table 2 values.

All environmental samples are analyzed for gross activities by proportional counter and technetium activity by liquid scintillation. Uranium concentrations in environmental samples are determined either by alpha spectrometry or ICP/MS. The minimum detectable activities/concentrations are comparable to those for effluent samples.

Laboratory quality control (QC) includes the use of a dedicated Chain of Custody facility, formal written procedures, NIST-traceable standards, matrix spikes, duplicate and replicate samples, check samples, and blind and double blind QC samples.

The X-710 Analytical Laboratory also participates in a wide variety of external control programs for both radiological and non-radiological parameters. These programs vary from time to time. Some of the programs that the Corporation has participated in over the years include: the EPA Discharge Monitoring Report Study, the NIOSH Proficiency Analytical Testing Program, the EPA Water Pollution Performance Evaluation Study, the EPA Water Supply Study, the NIOSH Environmental Lead Proficiency Analytical Testing Program, the Proficiency Environmental Testing program, a commercial program sponsored by the Analytical Products Group, Inc. of Belpre, Ohio, the DOE Environmental Measurements Laboratory (EML) Radionuclide Quality Assessment Program, and the DOE's Mixed Analyte Performance Evaluation Program (MAPEP).

Various matrix samples such as, water, air filters, soil, tissue, and vegetation have been analyzed for a wide variety of radioactive isotopes as part of the EML program.

The MAPEP Program has tested for a variety of metals and for activity of various radionuclides, including Cs-137, Co-57, Co-60, and ^{239}Pu, in soils and waters.

In addition, the laboratory has taken advantage of opportunities to participate in the DOE’s New Brunswick Laboratory Safeguards Measurement Evaluation Program for uranium purity in UF_6 and uranyl nitrate, as well as in the Regular European Interlaboratory Measurement Evaluation Program for uranium isotopic measurements.

9.2.2.6 Description of Status of Federal/State/Local Permits/Licenses

As described in Section 1.0 of the accompanying Supplement to the Environmental Report, the Licensee is in compliance with its Federal and State permits and licenses. There are no local permits or licenses applicable to the Lead Cascade.

9.2.3 Integrated Safety Analysis Summary

An Integrated Safety Analysis Summary for the Lead Cascade 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 summary is submitted for review (separate from this license application) as a document titled, Integrated Safety Analysis Summary for the American Centrifuge Lead Cascade Facility (U).

Table 9.2-1
PORTS Action Levels for Radionuclide Effluents

Weekly Sample Results		
Uranium[a]	**Technetium[a]**	**Required Actions[b]**
Baseline Effluent Quantity (BEQ)	Baseline Effluent Quantity (BEQ)	Review emission data for previous 6 months for trends, and estimate probable impact over calendar year. Evaluate whether additional emission controls would significantly reduce public exposure.
10 x BEQ or 2 x BEQ averaged over 6 months	80 x BEQ or 16 x BEQ averaged over 6 months	Determine whether increased emissions are ongoing or a single spike. Initiate investigation into cause(s) of increased emissions. Evaluate whether mitigative and/or corrective measures are necessary to reduce public dose. Implement mitigative and/or corrective measures as needed.
EPA Reportable Quantity[c] (RQ) (0.1 Ci in 24 hours)	EPA Reportable Quantity[c] (RQ) (10 Ci in 24 hours)	Notify Plant Shift Superintendent. Trace source of abnormal emissions and establish control or shutdown as needed. If emissions cannot be mitigated within 24 hours, elevate to next level.
1 Ci[d]	8 Ci[d]	Close affected vents until control of emissions is re-established.

a. Uranium has an approximately 8-fold greater dose rate response than ^{99}Tc over all air dominated exposure pathways.

b. Required actions for any level include all required actions listed under lower emission levels.

c. Reportable Quantity (RQ) does not include permitted emissions. PORTS is regulated under 40 CFR Part 61, Subpart H for release of airborne radionuclides from the entire plant site up to the equivalent of 10 mrem/year EDE to the most exposed member of the public.

d. 1 Ci or 8 Ci in one weekly sample analysis. Approximately equivalent to 2 mrem at most exposed residence.

Note: The Plant Shift Superintendent has the authority to allow a restart.

Table 9.2-2
BEQs for Lead Cascade Discharges

	Total Uranium (millicuries per week)
Vent	
Lead Cascade Process Vent	0.1 mCi/wk
Outfalls	
X-2230N West Holding Pond	$2.5x10^{-8}$ µCi/mL
X-2230M Southwest Holding Pond	$2.5x10^{-8}$ µCi/mL

Table 9.2-3
Maximum Anticipated Gaseous Discharges [a]

Discharge Point	**Total Uranium [b]**	
	µCi/mL [c]	mCi/wk [d]
X-3001 Process Vent	$0.97x10^{-16}$ to $2.1x10^{-16}$	0.1
10 CFR Part 20, App. B, Table 2	$3x10^{-12}$	------

[a] Based on Lead Cascade releases only.

[b] Since all uranium isotopes present at PORTS have the same discharge limit, all uranium isotope activities are lumped into a Total Uranium activity for simplify comparison to the Table 2 limits.

[c] Anticipated uranium concentration is based on engineering estimates of emissions and atmospheric dispersion modeling.

[d] By definition, anticipated activity discharges are less than the Baseline Effluent Quantity.

Table 9.2-4
Maximum Anticipated Liquid Discharges [a]

Discharge Point	Total Uranium [b]	
	µCi/mL [c]	mCi/wk [d]
X-6619 STP Discharge (NPDES Outfall 003)	<5x10^{-8}	<4.1
RCW Blowdown (NPDES Outfall 004)	1x10^{-8}	<1.4
X-2230N (NPDES Outfall 012)	<5x10^{-9}	<0.16
X-2230M (NPDES Outfall 013)	<8x10^{-9}	<0.13
10 CFR Part 20, App. B, Table 2	3x10^{-7}	------

[a] Based on all site contributions. Lead Cascade effluents are not expected to make a quantifiable difference in existing discharge levels.

[b] Since all uranium isotopes present at PORTS have the same discharge limit, all uranium isotope activities are lumped into a Total Uranium activity to simplify comparison to the Table 2 limits.

[c] Anticipated concentrations are based on measured concentrations in weekly composite samples from January 2007 through December 2009.

[d] By definition, anticipated activity discharges are less than the Baseline Effluent Quantity.

Table 9.2-5
PORTS Environmental Baseline Activities/Concentrations
Soil and Vegetation Results

	Total Uranium µg/g	Technetium pCi/g	Gross Alpha pCi/g	Gross Beta pCi/g
Soil / Onsite / 1999 - 2001				
Num. of Samples	70 (0)	70 (47)	70 (27)	70 (19)
Average	3.3	<0.2	<7.2	<13
Minimum	2.5	<0.1	<2.1	1.2
Maximum	4.4	1.1	13	23
Soil / Onsite / 2007 – 2009				
Num. of Samples	68 (0)	68 (62)	68 (42)	68 (10)
Average	2.7	<0.26	<6.00	<10.8
Minimum	1.6	<0.11	2.75	<6.18
Maximum	4.0	0.50	9.35	60.0
Soil / Offsite / 1999 – 2001				
Num. of Samples	45 (0)	45 (32)	45 (19)	45 (16)
Average	3.3	<0.2	<7.2	<15
Minimum	2.3	<0.1	<2.1	<8.0
Maximum	4.6	0.4	13	47
Soil / Offsite / 2007 – 2009				
Num. of Samples	50 (0)	50 (47)	50 (23)	50 (9)
Average	2.6	<0.28	<6.04	<9.63
Minimum	2.0	<0.22	<2.65	<5.97
Maximum	4.3	2.13	10.3	21.1
Soil / Remote / 1999 – 2001				
Num. of Samples	83 (0)	83 (77)	83 (39)	83 (27)
Average	3.3	<0.1	<7.4	<14
Minimum	2.1	<0.1	<3.3	<6.8
Maximum	5.9	0.8	16	21
Soil / Remote / 2007 – 2009				
Num. of Samples	80 (0)	80 (80)	80 (32)	80 (18)
Average	3.1	<0.25	<6.57	<12.3
Minimum	1.8	<0.22	2.80	6.44
Maximum	5.9	<0.32	12.9	62.9
Soil / Background / 1999 – 2001				
Num. of Samples	24 (0)	24 (20)	24 (8)	24 (13)
Average	3.8	<0.2	<7.6	<14
Minimum	2.2	<0.1	<4.5	<8.0
Maximum	6.8	0.5	16	25
Soil / Background / 2007 – 2009				
Num. of Samples	24 (0)	24 (23)	24 (11)	24 (7)
Average	3.2	<0.26	<5.72	<8.34
Minimum	2.1	<0.22	2.15	3.41
Maximum	6.5	0.37	8.18	12.9

Table 9.2-5
PORTS Environmental Baseline Activities/Concentrations
Soil and Vegetation Results

	Total Uranium µg/g	**Technetium pCi/g**	**Gross Alpha pCi/g**	**Gross Beta pCi/g**
Vegetation / Onsite / 1999 - 2001				
Num. of Samples	68 (66)	68 (55)	-----	-----
Average	<0.26	<0.4	-----	-----
Minimum	<0.25	<0.1	-----	-----
Maximum	0.9	7.3	-----	-----
Vegetation / Onsite / 2007 – 2009				
Num. of Samples	68 (37)	68 (63)	-----	-----
Average	<0.023	<0.30	-----	-----
Minimum	<0.012	<0.22	-----	-----
Maximum	0.139	3.09	-----	-----
Vegetation / Offsite / 1999 – 2001				
Num. of Samples	45 (45)	45 (34)	-----	-----
Average	<0.25	<0.3	-----	-----
Minimum	<0.25	<0.1	-----	-----
Maximum	<0.25	3.3	-----	-----
Vegetation / Offsite / 2007 – 2009				
Num. of Samples	50 (30)	50 (46)	-----	-----
Average	<0.017	<0.32	-----	-----
Minimum	<0.012	<0.22	-----	-----
Maximum	0.059	2.03	-----	-----
Vegetation / Remote / 1999 – 2001				
Num. of Samples	83 (81)	83 (73)	-----	-----
Average	<0.25	<0.2	-----	-----
Minimum	<0.25	<0.1	-----	-----
Maximum	0.25	0.5	-----	-----
Vegetation / Remote / 2007 – 2009				
Num. of Samples	80 (40)	80 (74)	-----	-----
Average	<0.025	<0.26	-----	-----
Minimum	<0.012	<0.17	-----	-----
Maximum	0.244	0.51	-----	-----
Vegetation / Background / 1999 – 2001				
Num. of Samples	24 (23)	24 (21)	-----	-----
Average	<0.25	<0.2	-----	-----
Minimum	<0.25	<0.1	-----	-----
Maximum	0.28	0.4	-----	-----
Vegetation / Background / 2007 – 2009				
Num. of Samples	24 (10)	24 (24)	-----	-----
Average	<0.021	<0.27	-----	-----
Minimum	<0.012	<0.22	-----	-----
Maximum	0.059	<0.35	-----	-----

Table 9.2-6
PORTS Environmental Baseline Activities/Concentrations
Surface Water and Sediment Results

	Total Uranium µg/L	Technetium pCi/L	Gross Alpha pCi/L	Gross Beta pCi/L
Surface Water / Upstream Scioto River / 1999 - 2001				
Num. of Samples	156 (3)	156 (150)	156 (123)	156 (82)
Average	<2.0	<15	<6	<13
Minimum	<1.0	<6	<2	<6
Maximum	32.6	28	12	40
Surface Water / Upstream Scioto River / 2007 – 2009				
Num. of Samples	158 (0)	158 (158)	158 (131)	158 (76)
Average	1.5	<9	<5.7	<7.0
Minimum	0.7	<7	<1.6	<3.8
Maximum	2.5	<10	12	17.8
Surface Water / Downstream Scioto River / 1999 - 2001				
Num. of Samples	156 (2)	156 (150)	156 (119)	156 (84)
Average	<1.7	<15	<6	<14
Minimum	<1.0	<6	<2	<7
Maximum	2.6	28	16	32
Surface Water / Downstream Scioto River / 2007 – 2009				
Num. of Samples	158 (2)	158 (158)	158 (139)	158 (86)
Average	<1.4	<9	<5.6	<6.7
Minimum	<0.1	<7	<2.6	<4.9
Maximum	2.6	<10	12.3	16.2
Surface Water / Background Creeks / 1999 – 2001				
Num. of Samples	144 (137)	144 (142)	144 (131)	144 (100)
Average	<1.0	<14	<4	<11
Minimum	<1.0	<6	<1	<6
Maximum	3.1	28	11	24
Surface Water / Background Creeks / 2007 – 2009				
Num. of Samples	128 (71)	128 (128)	128 (120)	128 (90)
Average	<0.2	<9	<4.2	<6.3
Minimum	<0.1	<8	<0.9	<4.6
Maximum	1.3	<9	8.3	49.8
	Total Uranium µg/g (dry wt)	**Technetium pCi/g (dry wt)**	**Gross Alpha pCi/g (dry wt)**	**Gross Beta pCi/g (dry wt)**
Sediment / X-2230M Southwest Holding Pond / 1999 – 2001				
Num. of Samples	6 (0)	6 (4)	6 (1)	6 (2)
Average	4.3	<0.2	<12	<11
Minimum	3.4	<0.1	<7	<7
Maximum	6.2	0.2	18	36
Sediment / X-2230M Southwest Holding Pond / 2007 – 2009				
Num. of Samples	6 (0)	6 (6)	6 (5)	6 (4)
Average	2.1	<0.25	<6.3	<7.2
Minimum	1.5	<0.22	<4.4	<6.7
Maximum	2.7	<0.27	8.0	7.7

Table 9.2-6
PORTS Environmental Baseline Activities/Concentrations
Surface Water and Sediment Results

	Total Uranium µg/g (dry wt)	Technetium pCi/g (dry wt)	Gross Alpha pCi/g (dry wt)	Gross Beta pCi/g (dry wt)
Sediment / X-2230N West Holding Pond / 1999 – 2001				
Num. of Samples	8 (0)	8 (2)	8 (2)	8 (7)
Average	3.4	<0.3	<8	<11
Minimum	2.9	<0.1	<5	<7
Maximum	4.9	0.6	10	17
Sediment / X-2230N West Holding Pond / 2007 – 2009				
Num. of Samples	6(0)	6 (5)	6 (3)	6 (5)
Average	2.3	<0.31	<6.5	<8.0
Minimum	1.1	<0.23	<3.5	<6.2
Maximum	4.2	0.59	10.0	11.6
Sediment / Upstream Scioto River / 1999 – 2001				
Num. of Samples	7 (0)	7 (7)	7 (4)	7 (6)
Average	2.1	<0.1	<6	<12
Minimum	0.9	<0.1	<3	<6
Maximum	4.6	<0.2	9	24
Sediment / Upstream Scioto River / 2007 – 2009				
Num. of Samples	16 (0)	16 (15)	16 (13)	16 (10)
Average	1.8	<0.31	<6.3	<7.9
Minimum	1.2	<0.23	<4.9	<6.3
Maximum	2.5	1.02	9.3	14.6
Sediment / Downstream Scioto River / 1999 – 2001				
Num. of Samples	6 (0)	6 (5)	6 (4)	6 (4)
Average	3.1	<0.2	<8	<14
Minimum	2.2	<0.1	<6	<10
Maximum	4.4	0.4	10	19
Sediment / Downstream Scioto River / 2007 – 2009				
Num. of Samples	10 (0)	10 (10)	10 (3)	10 (4)
Average	1.8	<0.25	<5.9	<8.1
Minimum	1.4	<0.22	3.3	<6.3
Maximum	2.3	<0.28	8.4	12.5
Sediment / Background Creeks / 1999 – 2001				
Num. of Samples	24 (0)	24 (22)	24 (15)	24 (16)
Average	3.2	<1.4	<7	<12
Minimum	1.3	<0.1	<4	<8
Maximum	6.7	9.9	11	17
Sediment / Background Creeks / 2007 – 2009				
Num. of Samples	24 (0)	24 (24)	24 (12)	24 (16)
Average	2.2	<0.25	<6.0	<8.1
Minimum	0.8	<0.22	<3.4	<6.4
Maximum	4.35	<0.28	10.6	13.1

Table 9.2-7
Environmental Baseline Radiation Levels

Area of Readings	Average	Minimum	Maximum
1999-2001			
Onsite	10.4 µrem/hr	6.6 µrem/hr	17.9 µrem/hr
X-746 Cylinder Yard	71.5 µrem/hr	60.1 µrem/hr	82.3 µrem/hr
Boundary	10.5 µrem/hr	6.2 µrem/hr	22.6 µrem/hr
Piketon	9.6 µrem/hr	7.4 µrem/hr	13.9 µrem/hr
Camp Creek	10.2 µrem/hr	7.8 µrem/hr	14.9 µrem/hr
2007-2009			
Onsite	12.7 µrem/hr	6.8 µrem/hr	24.1 µrem/hr
X-746 Cylinder Yard	52.7 µrem/hr	64.1 µrem/hr	92.7 µrem/hr
Boundary	11.8 µrem/hr	8.3 µrem/hr	15.6 µrem/hr
Piketon	11.6 µrem/hr	9.4 µrem/hr	13.9 µrem/hr
Camp Creek	11.7 µrem/hr	9.2 µrem/hr	14.9 µrem/hr

The information within this figure has been determined to contain Export Controlled Information and is located in Appendix B of this license application

Figure 9.2-1
Locations of USEC-Leased Outfalls
Discharging to Waters of the United States

9.3 Reports to the Nuclear Regulatory Commission

The Lead Cascade uses full-scale equipment and laboratory samples are withdrawn to obtain information on American Centrifuge enrichment technology. The cascade is operated on recycle where the enriched product stream is recombined with the depleted stream prior to being re-fed to the cascade. No enriched product withdrawals are made from the Lead Cascade for commercial use; however, small samples are taken for laboratory analysis. Accordingly, the effluent monitoring reporting requirements specified in 10 CFR 70.59 are not applicable to the Lead Cascade. However, the Lead Cascade is committed to the NRC, under the PORTS Environmental Protection Program described in USEC-02, to provide, as requested, periodic reports generated under EPA regulations that cover all radioactive effluents from the site. The Lead Cascade effluents are a small fraction of the total site effluents. Lead Cascade effluents are incorporated into the existing reports, described below. These reports are available upon request for inspection at the facility.

9.3.1 Gaseous Effluents

The Corporation is required under 40 CFR 61.94 to submit a written report to the EPA by June 30 of the each year detailing: site operations and gaseous effluent monitoring during the previous year, gaseous radioactive effluents over the previous year, an assessment of the public EDE caused by those effluents, and an explicit comparison of the calculated EDE to the EPA public dose limit (10 mrem annually). The EDE data in this report is required by EPA to incorporate the public dose due to both the Corporation's and DOE operations, even though DOE and the Corporation submit separate reports for their own operations. This report would be used monthly if the maximum public EDE exceeds 10 mrem annually.

9.3.2 Liquid Effluents

The Corporation is required under Condition II.K of the NPDES Permit to submit a written report to the Ohio EPA within 60 days of the end of each calendar quarter summarizing measured radionuclide concentrations and activity loadings in all outfalls to unrestricted areas. The report consists of a table for each covered outfall giving the measured concentrations for each weekly composite sample followed by the loading, in millicuries, calculated from the concentrations and the measured daily flow. (The daily flow itself is reported to the Ohio EPA separately in a Monthly Operating Report.) The foot of each table lists the quarterly totals, means, maximums and minimums as consistent with the Ohio EPA's guidance for the Monthly Operating Report.

9.3.3 Direct Radiation and Compliance Status

The Corporation compiles an annual report summarizing the direct radiation monitoring results and any Notices of Violation, permit exceedances, or other citations (other than those from NRC itself) over the last calendar year.

9.3.4 Baseline Effluent Quantity Reports

The Corporation is required by the Environmental Protection Program established in USEC-02 to quantify weekly radioactive effluents and respond as described in Table 9.2-1 to any weekly effluent that exceeds the action levels in that table. Many years of experience by the Corporation and its predecessor organizations have shown that radioactive effluent levels less than the action levels in Table 9.2-1 do not produce a public radiation dose that is within an order of magnitude of the dose constraint in 10 CFR 20.1101, let alone the dose limit of 10 CFR 20.1301. Any weekly effluent levels that exceed the action levels in Table 9.2-1 require a written estimate of the probable impact of the effluent, in conjunction with all other monitored effluents from the Licensee operations, on the annual public radiation dose.

These reports are available upon request by the NRC. They are not routinely submitted to outside authorities because they are considered interim assessments that are superseded by the annual public dose assessment and quarterly reports described in Sections 9.3.1 and 9.3.2 of this license application.

9.4 References

1. 40 *Code of Federal Regulations* Part 61, National Emission Standards for Hazardous Air Pollutants

2. 40 *Code of Federal Regulations* Part 266, Standards for the Management of Specific Hazardous Wastes and Specific Types of Hazardous Waste Management Facilities

3. Environmental Report for the American Centrifuge Lead Cascade Facility

4. Integrated Safety Analysis Summary for the American Centrifuge Lead Cascade Facility (U)

5. *National Environmental Policy Act* of 1969

6. NRC Regulatory Guide 1.109, Revision 1, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I*, October 1977

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

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

9. Draft NUREG-1748, *Environmental Review Guidance for Licensing Actions Associated with NMSS Program.*

10. *Resource Conservation and Recovery Act* of 1976

11. USEC-02, Application for United States Nuclear Regulatory Commission Certification, Portsmouth Gaseous Diffusion Plant

12. Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,* September 1988

13. LA-3605-0001, *License Application for the American Centrifuge Plant in Piketon, Ohio*

14. LA-2605-0002S, *Supplement to the Environmental Report for the American Centrifuge Lead Cascade Facility*

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11.0 MANAGEMENT MEASURES

Management measures are functions, performed by the Licensee that are applied to items relied on for safety (IROFS) to provide reasonable assurance that the IROFS are available and able to perform their functions when needed for the Lead Cascade. The phrase "available and reliable," as used in 10 *Code of Federal Regulations* (CFR) Part 70, means that, based on the analyzed, credible conditions in the Integrated Safety Analysis (ISA), IROFS will perform their intended safety function when needed to prevent accidents or mitigate the consequences of accidents to an acceptable level. Management measures are implemented to provide reasonable assurance of compliance with the performance requirements, considering factors such as necessary maintenance, operating limits, common-cause failures, and the likelihood and consequences of failure or degradation of the IROFS and the measures. This chapter addresses each of the management measures included in the 10 CFR Part 70 definition of management measures, i.e., configuration management (CM), maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance (QA) elements. The degree to which measures are applied to the IROFS is a function of the item's importance in terms of meeting the performance requirements as evaluated in the ISA.

The descriptions of management measures address in sufficient detail how the measure is designed, organized, and conducted to enable the reviewer to understand the capability of the measure to be implemented at the Lead Cascade.

11.1 Configuration Management

The CM Program for the Lead Cascade is described in the following paragraphs.

11.1.1 Configuration Management Policy

In accordance with 10 CFR 70.72, a CM Program is implemented to ensure that changes from the Lead Cascade Baseline Configuration are identified and controlled to help ensure safety through consistency among the facility design and operational requirements, the physical configuration, and the facility documentation. The CM Program includes:

- Identification and documentation of IROFS;
- Organizational descriptions of duties and responsibilities; and
- Administrative controls, procedures and policies, to implement and document activities that maintain the facility's configuration.

The goal of the CM program is to ensure that the facility has accurate, current documentation that matches the facility's physical/functional configuration, while complying with applicable requirements.

11.1.1.1 Program Overview

The Technical Services Manager has primary responsibility for the implementation of the CM Program for the Lead Cascade. The CM Program is applicable to the facility, structures, processes, systems, equipment, components, computer programs, and activities of personnel, regardless of the item's Quality Level (QL) classification.

CM Program procedures provide for a graded application of resources taking into consideration:

- QL (risk significance);
- Applicable regulations, industry codes, and standards;
- Complexity or uniqueness of an item or activity and the environment in which it has to function;
- Quality history of the item in service;
- Degree to which functional compliance can be demonstrated or assessed by test, inspection, or maintenance methods;
- Anticipated life span;
- Degree of standardization;
- Importance of data generated;
- Reproducibility of results; and
- Consequence of failure.

QLs are established in accordance with their importance to safety as follows:

Level	**Criteria**
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.

The CM Program implementing procedures provide a management system to evaluate, implement and track each change to the facility, structures, processes, systems, equipment, components, computer programs, and activities of personnel. Procedures are utilized to ensure that the following items are addressed, in accordance with 10 CFR 70.72(a)(1) – (6), prior to implementing any change:

- The technical basis for the change;
- Impact of the change on safety and health or control of licensed material;
- Revisions, if required, to existing operating procedures, including any necessary training or retraining before operation;
- Authorization requirements for the change;
- For temporary changes, the approved duration (i.e., expiration date) of the change; and
- The impacts or modifications to the ISA, ISA Summary, or other safety program information that is part of this application.

11.1.1.2 Key Program Responsibilities

The following responsibilities are identified by the responsible Lead Cascade manager and functional area. Personnel from the Portsmouth Gaseous Diffusion Plant (PORTS) fulfill a number of the functions.

- **Technical Services Manager**
  - Manages the CM Program.
  - Is the Facility Design Authority (FDA) responsible for:
    - ➢ Establishing the design requirements;
    - ➢ Ensuring design output information (documents and data) appropriately and accurately reflects the design input; and
    - ➢ Maintaining the facility's ISA and ISA Summary.

- Performs design/modification processes that implement the design control and design change control requirements established in the Quality Assurance Program Description (QAPD), which includes controls for design inputs, design verification (including analysis software), design changes, design interfaces and design documentation and records.

- Manages the Temporary Change Process.

- Identifies and defines IROFS as part of the ISA process.

- Performs reviews of facility changes in accordance with the requirements of 10 CFR 70.72.

- Establishes inspection and acceptance criteria for IROFS.

- Ensures that appropriate documents and procedures are updated to be consistent with modifications.

- Issues the documentation that defines boundaries for IROFS in the CM Program.

- Establishes and maintains a controlled database for IROFS information.

- Assists in work package preparation and identification of post-maintenance test requirements to assure that the critical design characteristics of IROFS are satisfied.

• **Procurement Manager**

- Develops procedures in accordance with the QAPD for procurement and control of items.

- Purchases IROFS and replacement parts only from authorized vendors and in accordance with the requirements and technical specifications as identified by the Technical Services Organization.

• **Operations Manager**

- Ensures modifications are not made to a design or operational configuration without proper review and approval.

- Performs and documents operational, post-maintenance tests/checks, and post-modification tests to assure IROFS are operating as intended.

- Issues work orders or other authorizations prior to maintenance, testing, or modification activities.

- Records the occurrence of tests, calibrations and maintenance activities.

- Ensures approved procedures are used for operations involving the replacement or adjustment of IROFS.

- **Maintenance Manager**

  - Develops and implements procedures to execute a work control process which provides for:

    - ➢ Verification of data, performance or documentation where specified by the FDA; and

    - ➢ Documentation of material used to ensure design specifications are met.

  - Ensures maintenance personnel are knowledgeable of requirements for working on IROFS systems/items.

  - Performs work on IROFS only after receiving issuance of an approved maintenance work package.

  - Ensures modifications are not made to a design or operational configuration without proper review and approval.

  - Identifies and transmits completed work packages for IROFS to Records Management and Document Control (RMDC) in a timely manner.

  - Ensures that only accepted IROFS are stored and issued for work.

  - Maintains items in a manner that complies with engineering issued requirements.

Maintenance is described in Section 11.2 of this license application.

- **Training Manager**

  Procedures

The Procedures process is described in Section 11.4 of this license application. A procedures control program is utilized to ensure technical, operations, maintenance, and administrative procedures used to apply the CM Program processes are properly developed, reviewed, approved, revised, and controlled.

Training

- Provides technical training support to Lead Cascade personnel who are relied upon to operate, maintain, or modify IROFS.

- Provides training support to Technical Services, Operations, and Maintenance personnel to ensure training is updated as a result of changes to the facility.

Training and Qualification is described in Section 11.3 of this license application.

Records Management and Document Control

- Develops and operates a RMDC program that controls and issues designated documents and acts as the repository with retrieval capabilities for controlled documents and records necessary to maintain the facility's design history.

- Maintains an index of documents and software that are required to be controlled.

RMDC is described in Section 11.7 of this license application.

- **Quality Assurance Manager**

  - Assists in the development and implementation of the acceptance process to assure that the critical design characteristics are satisfied for non-commercial grade IROFS.

  - Assists in the acceptance process for commercial grade IROFS.

  - Verifies that FDA supplied acceptance criteria are met and that accepted items are appropriately identified.

  - Establishes a program for in-process inspection of maintenance work packages in accordance with acceptance criteria contained in maintenance procedures or provided by the FDA to assure that the critical design characteristics of IROFS are satisfied.

  - Conducts audits and surveillances of processes that implement the CM Program, as specified by the QAPD.

  - Audits vendors and suppliers in accordance with the QAPD.

11.1.2 Design Requirements

- Design requirements are developed to support safety functions, environmental impact-oriented functions, and mission-based functions.

  - IROFS are identified in the ISA Summary. Design requirements for IROFS or for other systems or components required to support the ISA and/or meet the performance requirements established in 10 CFR 70.61 are developed in accordance with 10 CFR 70.64.

  - Other systems or components to support environmental impact-oriented functions and mission-based functions are identified in System Requirements Documents (SRDs).

- The design requirements to support the IROFS and other systems or components are developed by the Technical Services Organization and documented in Design Criteria Documents for each facility/system. Prior to approval, these documents are reviewed to determine their adequacy, accuracy, and completeness.

- The FDA approves Design Criteria Documents.

- After approval by the FDA, the Design Criteria Documents and the ISA Summary, as well as Design Basis Documents, Lead Cascade SRDs, and as-built drawings and specifications, provide the baseline configuration for the facility.

- Changes to any design basis or design requirements are modifications that are controlled by the change control process described in Section 11.1.4 of this license application.

- The Design Criteria Documents are controlled documents. When modifications result in changes to these documents, the changes are controlled in accordance with the RMDC requirements described in Section 11.7 of this license application.

11.1.3 Document Control

Procedures, documents, and records control programs provide for centralized control and issuance of documents necessary for the maintenance of the Lead Cascade configuration and provide a repository for records to verify this maintenance. RMDC requirements are described in Section 11.7 of this license application.

11.1.3.1 Procedures

The procedure control program assures that procedures are generated, reviewed, approved, and distributed in a controlled manner. Section 11.4 of this license application describes the procedure control program.

11.1.3.2 Records Management and Document Control

A document control program ensures that changes to approved and controlled documents are:

- Issued in a timely manner;
- Distributed to controlled copy holders; and
- Maintained available to support daily work activities.

Controlled documents, in support of the CM Program, are identified in the procedures that generate the documents. RMDC personnel maintain an index of documents that are required to be controlled. The documents include, but are not limited to, such documents as:

- Procedures addressing activities affecting IROFS
- Design documents (e.g., drawings, analyses, and calculations)
- The IROFS database change records
- Engineering specification data sheets, which include the technical requirements, vendor data requirements, and the commercial grade dedication requirements
- The ISA Summary and other hazard analyses
- Procedures and plans addressing emergency operating and response plans
- Records to support maintenance and verification of the Lead Cascade configuration such as:
  - Design modification packages
  - Acceptance records for receipt of material, shop and field inspection of work processes supporting maintenance, repair, and testing records
  - Maintenance, repair, and modification construction and installation work packages
  - Documentation used by Operations to record verification and test data

The RMDC Program is described in Section 11.7 of this license application.

11.1.4 Change Control

In accordance with 10 CFR 70.72, the Licensee may make changes to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel, without prior U.S. Nuclear Regulatory Commission (NRC) approval, if the change:

- Does not:
  - Create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 and that have not previously been described in the ISA Summary; or
  - Use new processes, technologies, or control systems for which the licensee has no prior experience.
- Does not remove, without at least an equivalent replacement of the safety function, an IROFS that is listed in the ISA Summary;
- Does not alter any IROFS, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR 70.61; and
- Is not otherwise prohibited by 10 CFR 70.72, a license condition, or an NRC order.

In accordance with the requirements of 10 CFR 70.72, the Licensee implements change control processes for changes to the physical facility and for changes to procedures and controlled documents. These processes are described in Sections 11.1.4.1 and 11.1.4.2 of this license application, respectively. The Facility Safety Review Committee reviews appropriate changes to the facility or facility's operations, including tests and experiments, as specified in procedures. Procedures also specify the approval authority for the changes.

11.1.4.1 Control of Changes to the Physical Facility

The Licensee has implemented a change control process that is implemented through written procedures to control changes to the physical facility. This change control process meets the requirements established in 10 CFR 70.72 and in the QAPD. Key elements of the change control process are described in the following paragraphs:

- Requests for engineering assistance, after initiator's management approval, are forwarded to the FDA for:
  - Review to determine if the proposed change is acceptable based upon scope, applicability, justification, and/or technical merit;
  - Engineering approval; and

  - Disposition and assignment to the appropriate engineering discipline.

- Facility modifications, additions, or changes have a 10 CFR 70.72 review performed to determine if the change can be made without prior NRC approval. Information utilized in the 10 CFR 70.72 review includes the following, as appropriate:

  - SRDs;

  - Conceptual design descriptions;

  - Drawings/specifications;

  - Other documentation providing a project description; and

  - Changes in the project.

- Modifications (permanent and temporary) are evaluated, as appropriate, for any required changes or additions to the facility's procedures, personnel training, testing programs, or the ISA Summary. Modifications are also evaluated, as appropriate, for potential radiation exposure, nuclear criticality safety, and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include: modification costs, similar completed modifications, quality assurance aspects, potential equipment availability or maintainability concerns, constructability concerns, environmental considerations, and human factors.

- Critical repair parts for IROFS are identified during the design process.

- Proposed facility modifications receive an independent, technical review that considers the technical feasibility and merit of the proposed change and the identification of appropriate interfaces for inclusion in the change package (e.g., procedures, training, safety).

A final review prior to release for operation is conducted which verifies that:

  - The safety analysis documentation is complete and approved;

  - Operational procedure changes, if required, are completed and other supporting procedure changes have been initiated;

  - Operational training and qualification changes, if required, have been completed;

  - Design changes are completed and any as-built changes are identified and approved;

  - Document changes, if required, are completed;

- For temporary changes, the change duration is documented and the modified equipment tagged;

- Post-modification testing has been successfully completed; and

- All appropriate approvals have been obtained.

11.1.4.2 Control of Changes to Procedures and Controlled Documents

Changes to procedures and controlled documents are controlled in accordance with the programs described in Section 11.4, Procedures, and Section 11.7, Records Management of this license application, respectively.

11.1.5 Assessments

The CM Assessment Program systematically evaluates the development and effective implementation of the CM Program processes. It assesses the adequacy of the implementation of administrative requirements, the configuration of items, and their documentation. The CM Assessment Program includes both initial and periodic assessments. Both document assessments and physical assessments (system walkdowns) are conducted periodically to confirm the adequacy of the CM function.

Initial assessments of the CM program are performed prior to operation of the Lead Cascade. The initial assessment provides for field verification of design requirements and design documentation, verification of procedures, and verification of training.

Periodic assessments of the CM Program are performed as part of the commitments contained in Section 11.5, Audits and Assessments, of this license application, and the QAPD.

Any deficiencies or recommendations for programmatic improvements are identified, documented, and addressed in accordance with the requirements established in the Lead Cascade's Corrective Action Program, described in Section 11.6, Incident Investigations, of this license application.

11.1.6 Design Verification

The structures for the Lead Cascade Project were built by the U.S. Department of Energy (DOE) for the Gas Centrifuge Enrichment Plant program and are leased by the Licensee. New structures were not constructed to house the Lead Cascade. A small portion of the available leased space is utilized for Lead Cascade operations. Existing facilities and equipment were also used to fill many of the design requirements.

The Licensee is utilizing a process to verify that the design of the existing facilities and equipment meet the Lead Cascade Design Requirements. This section briefly describes the process for design verification and incorporation of existing design information into the Lead Cascade Baseline Configuration. The process is depicted in Figure 11.1-1 and described below.

- The required existing structures, systems, and components (SSCs) are identified and SSC Assessment Plans are prepared.

- Physical system walk-downs are then performed in accordance with the guidance provided in the SSC Assessment Plan to:

    - Compare the configuration of the SSC with original drawings, construction specifications, and procedures; and

    - Determine the current material condition of the SSCs.

- An Assessment Report is then prepared to describe the current configuration and condition of the SSC and to identify discrepancies for the as-built configuration or condition.

- The configuration and condition of the SSC, as documented in the Assessment Report, is then compared with the Lead Cascade SRDs.

- If it is determined through evaluation and validation that the existing SSC fulfills the requirements established by the SRD, the SSC is accepted and baseline configuration information for the SSC is incorporated into the Lead Cascade Baseline Configuration.

- If it is determined that the existing SSC does not fulfill the requirements established by the SRD, the differences are documented.

- Design criteria are then developed for the SSC and a design change package is prepared to perform the changes necessary to meet the Lead Cascade design criteria.

- After construction activities are performed to make the design changes, drawings, and specifications are as-built.

- The Design Criteria and as-built documentation are then incorporated into the Lead Cascade Baseline Configuration.

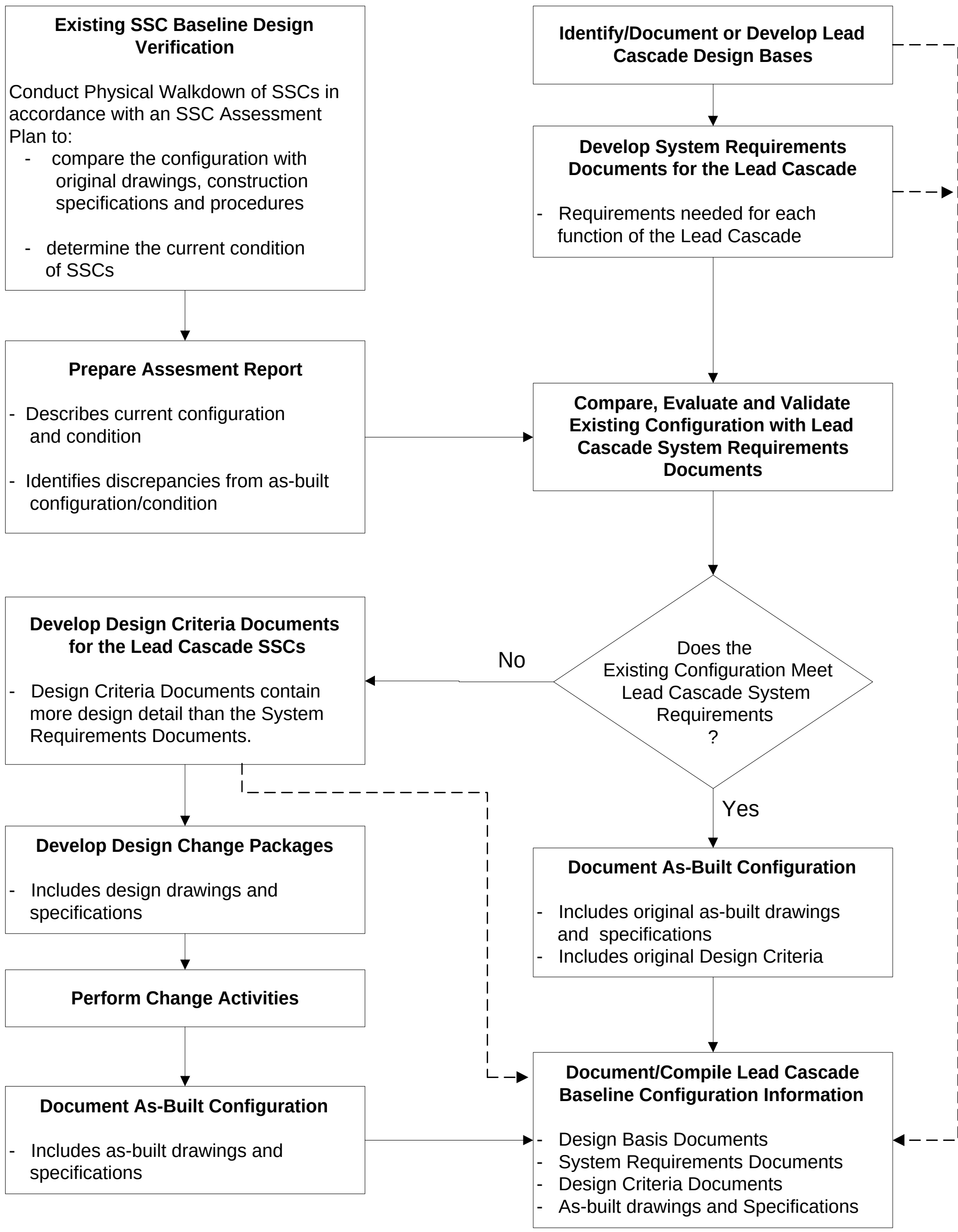

Figure 11.1-1
Lead Cascade Design Verification Process

11.2 Maintenance

The Maintenance Organization provides reliable and cost-effective maintenance of Lead Cascade equipment. Maintenance programs related to corrective and preventive maintenance are established to provide a level of inspection, calibration, repair, replacement, and testing that ensures each IROFS will be available and reliable to perform its intended function.

11.2.1 Maintenance Organization and Administration

The Maintenance Organization has policies, procedures, and programs that establish requirements and standards related to maintenance of facility equipment. These policies, procedures, and programs address:

- Personnel qualification and training
- Design/work control
- Corrective maintenance
- Preventive maintenance
- Surveillance/monitoring
- Post-maintenance testing
- Control of measuring and test equipment
- Equipment/work history

These requirements and standards are established for compliance with the quality assurance and configuration management programs. Effective implementation and control of maintenance activities are achieved through application of these standards that are periodically reviewed and assessed for compliance.

The Maintenance Manager is responsible for the overall coordination and management of the organization to provide safe and efficient performance during maintenance of facility equipment.

Maintenance Work Center Supervisors report to the Maintenance Manager. The Maintenance Work Center Supervisors are responsible for directing the activities of the Maintenance Shift Supervisors in the performance of preventive, predictive, and corrective maintenance and to provide support services on facilities and equipment, with the exception of centrifuge machines.

Shift Maintenance Supervisors, who report to the Maintenance Work Center Supervisors, are responsible for execution of maintenance on facility equipment. These responsibilities include:

- Supervision of craft personnel
- Coordination with support groups

- Ensuring that maintenance activities are appropriately planned in accordance with the work control process
- Qualification of personnel assigned to perform maintenance on Lead Cascade equipment
- Review of work practices by craft for compliance with maintenance and facility safety procedures

Craft personnel are responsible for:

- Compliance with Lead Cascade safety procedures while performing maintenance
- Compliance with maintenance procedures while performing maintenance
- Completion of documentation related to the maintenance activity

11.2.2 Personnel Qualification and Training

The selection and qualification of personnel in the Maintenance Organization is documented and implemented through procedures. Qualification requirements are established for craft maintenance positions.

Qualification requirements for craft positions are established specific to each classification. The level of knowledge of each candidate in the related field is described in Section 11.3.2 of this license application. Employees are required to successfully complete classroom and on-the-job training programs. An analysis of the responsibilities of each classification is performed to establish the content and type of training required for the position. This review considers each of the activities performed by each classification and the importance of that activity to safe operation of the facility and maintenance of IROFS. Consideration is also given to the complexity of the activity, the frequency of the activity must be performed by maintenance personnel, and the consequences if an error is made during the evolution. Skill-of-the-craft and availability of procedures or other approved technical document that direct performance of the maintenance evolution is also considered as part of this task analysis.

Contractors that work on or are performing activities that could affect IROFS follow the same maintenance guidelines as Lead Cascade maintenance personnel. In addition, a member of the Lead Cascade organization provides oversight of contractor activities.

11.2.3 Design/Work Control

Maintenance of Lead Cascade equipment is performed in a manner that maintains the documented configuration of facility systems. Prior to modification of systems, it is necessary to complete actions required by 10 CFR 70.72. A work control process establishes the necessary control, review, and approval process to maintain the documented configuration of Lead Cascade systems.

The need for maintenance is identified when an equipment owner initiates a request for work or by the automatic generation of preventive maintenance (PM) tasks or surveillances. The activity described by the request is evaluated to determine the class of work specified for the item requiring maintenance. Lead Cascade equipment is also classified to a specific QL by the Technical Services Organization. QLs are established in accordance with the equipment's relation to safety. Additional information regarding the graded approach taken to determine the QL of an item is found in Section 11.1 of this license application and in Section 2 of the QAPD.

The QL of an item requiring maintenance establishes the level of planning, extent of reviews, and approval required to perform the maintenance task. A work package is developed to direct and document maintenance activities involving a QL item. These work packages contain, as a minimum, a task description, approved work instructions or procedure, post-maintenance tests and equipment history documentation. The package contents may also include equipment drawings, vendor manuals, and safety permits. Compensatory actions are established prior to an IROFS being removed from service for maintenance.

Minor maintenance may be performed on equipment classified as QL-3. Such activities can normally be considered within the skill and training of the craft. These minor maintenance activities do not require work instructions, procedures, or development of a work package. A work package is developed when the maintenance activity would result in a change to the configuration of the system or for a complex evolution, even though working on a non-safety system.

The planning process addresses support required of other Lead Cascade organizations. The repair and/or replacement of IROFS are performed with like-for-like parts. Modifications to Lead Cascade systems may only be performed following evaluation and approval of the Technical Services Organization.

The work package to perform the maintenance activity is reviewed and approved by the appropriate disciplines. Appropriate technical and safety reviews and approvals are performed. At a minimum, review and approval of a representative from maintenance and the equipment owner is required before a work package can be used to perform maintenance on Lead Cascade equipment. The Technical Services Organization is required to review and approve work packages created for maintenance of QL items and packages developed for modification of Lead Cascade systems.

Maintenance activities are scheduled through an established work control process. The equipment owner establishes priorities for maintenance in his/her area of responsibility. A schedule is created and published which establishes a date for execution of the maintenance activity. The work is scheduled in advance to accommodate completion of the planning process. The process accommodates emergent, high priority work. Operations authorizes the performance of maintenance and removal of IROFS from service. It is also responsible for ensuring safe operations during removal of IROFS from service, including establishing any necessary compensatory measures. Operations is notified upon completion of maintenance.

The work control process provides configuration control of Lead Cascade equipment. This process requires an evaluation for availability of:

- Qualified personnel to perform the maintenance
- Approved work instructions and/or procedures
- Approved parts or substitutes
- Drawings
- Safety permits

Other documentation related to the maintenance activity may be included in the package as determined by the Maintenance Shift Supervisor to support the process.

11.2.4 Corrective Maintenance

Corrective maintenance is the action to check, troubleshoot, and repair equipment that has degraded or failed. The identification, prioritization, planning and scheduling of corrective maintenance activities are accomplished following the work control process described in Section 11.2.3. Corrective actions are performed to promptly remediate unacceptable performance deficiencies in IROFS and to eliminate or minimize the recurrence of these deficiencies.

11.2.5 Preventive Maintenance

PM is the activity performed on a periodic basis to prevent failures, facilitate performance, and maintain or extend the life of equipment. PMs help ensure that IROFS are available to perform their function and are reliable. The bases for PM tasks are developed through a review of manufacturer recommendations, available industry standards, and historical operating information, where available. The rationale for any deviations from industry standards or manufacturer's recommendations is documented. PMs are included in the work control process to facilitate planning, scheduling, and execution of these tasks.

Establishment of a PM task is coordinated by engineering and maintenance and requires input from various disciplines within the Technical Services Organization, as well as operations and maintenance personnel, as appropriate. The formal documented bases for the tasks are developed, evaluated, and approved by the Technical Services Organization. PM tasks may be changed, new tasks added or deleted, and recommendations made by operations, maintenance, or engineering personnel. Changes to tasks may be warranted as a result of a review of a system's performance. Feedback from PM, corrective maintenance, and incident investigations is used, as appropriate, to modify the frequency or scope of a PM activity. Specifically, preventive measures to alleviate premature failure may be added to the PM activity, or a reduction in frequency of a particular PM due to as-found conditions indicating that the PM is occurring more often than necessary, may be initiated.

11.2.6 Surveillances

Surveillance and monitoring at specified intervals are performed to verify the proper operation of IROFS and to measure the degree to which IROFS meet performance specifications. These surveillances are in the form of performance checks, calibrations, tests and/or inspections. The ISA Summary identifies the IROFS that are credited to be available and reliable to perform their design function for mitigation of credible events. The surveillance program provides a periodic check of the ability of these IROFS to perform their design safety function when called upon to do so. The Surveillance program design adheres to the 10 CFR 70.64 Inspection, Testing, and Maintenance Baseline Design Criteria (BDC).

Surveillances are included in the work control process to permit timely planning, scheduling, establishment of system or facility conditions, execution of the activity, and creation of documentation that identifies the results of the surveillance. The established frequencies are determined by the IROFS' degree of safety importance. The results of surveillance activities are trended to support the determination of performance trends for IROFS. When indicated by potential performance degradation, preventive frequencies are adjusted or other corrective actions taken, as appropriate.

11.2.7 Post-Maintenance Testing

A post-maintenance testing (PMT) program is established to provide assurance an IROFS will perform its intended function following maintenance activities. This test confirms that the maintenance performed was satisfactory, the identified deficiency has been corrected, and the maintenance activity did not adversely affect the reliability of the IROFS. This test is performed with acceptable results prior to return of the equipment for service.

PMT requirements are developed and included in work packages during the work planning process. The Technical Services Organization may provide support to the Operations and Maintenance Organizations in identifying PMT requirements. The PMT meets applicable codes and technical requirements and specify acceptance criteria. The results of the PMT are documented and retained in the work package with other documentation generated during the maintenance evolution.

11.2.8 Control of Measuring and Test Equipment

Maintenance programs include control of measuring and test equipment (M&TE) used during maintenance of Lead Cascade equipment. This program requires M&TE to be properly controlled, calibrated and adjusted, if necessary, at specified periods. The following are elements of the M&TE control program:

- M&TE is assigned a unique identifier
- Calibration intervals are defined
- M&TE is labeled to identify calibration/certification status

- An M&TE inventory is maintained

- M&TE determined to be out of tolerance during calibration is identified and an investigation conducted of equipment use since the previous calibration

- Calibration records are retained

- Control and storage requirements are defined for M&TE

Standards used for calibration of M&TE have the required accuracy, range and stability for the application. These standards are certified and traceable to the National Institute of Standards and Technology. If no national standard exists, the bases for calibration is documented and approved by the Technical Services Organization.

Additional requirements and standards are established as necessary to ensure compliance with Section 12 of the QAPD.

11.2.9 Equipment/Work History

Maintenance programs include data collection in the work control process. Maintenance on an IROFS requires the preparation of a work package that contains an equipment history form. This form is used to collect information from the craft personnel that are performing PM and corrective maintenance activities on an IROFS. The work package also contains a work-in-progress log used to document actions taken during the maintenance evolution. This documentation provides information regarding the as-found condition of an IROFS. This data is intended to identify the need for modifications and improvements for the maintenance program, to improve the reliability of an IROFS, and to ensure maintenance personnel are devoting their efforts to activities important to safety.

The information obtained from work packages is retained in a database for historical reference. The Technical Services Organization uses this database to evaluate the reliability of IROFS. This data, in addition to other indicators (e.g., results of incident investigations, the review of failure records required by 10 CFR 70.62(a)(3), and identified root causes) of item performance allow for a thorough review to determine if modifications to a system or a change in the maintenance program is necessary to ensure that IROFS are reliable and available when called upon. The actual documentation generated at the time of the maintenance evolution is retained in the work package and is controlled according to RMDC program practices.

11.3 Training and Qualifications

Those personnel who perform activities relied on for safety have the applicable knowledge and skills necessary to design, operate, and maintain the facility in a safe manner. They are tested as necessary to ensure that they are qualified on practices important to public and worker safety, safeguard of licensed material, and protection of the environment.

11.3.1 Training Program Organization and Administration

The Training Manager is responsible for preparation, presentation, and documentation of employee orientations, and for technical and qualification training program development and implementation. The Training Manager utilizes instructional technologists, technical trainers, and administrative support personnel to provide training program services. These training personnel interface with the Lead Cascade functional line managers to coordinate and assist in the design, development, implementation, and evaluation of the training and qualification programs in the following functional areas:

- **General Employee Training** (GET) for persons who require unescorted access (Section 11.3.4), including visitors (Section 11.3.5)

- **Radiation Worker Training** for personnel whose job requires them to have unescorted access to radiological restricted areas (Section 11.3.6)

- **Nuclear Criticality Safety Training** for personnel who handle or manage the handling of fissile material and work within Fissile Material Operations Areas (Section 11.3.7)

- **Environmental, Safety, and Health Training** for those persons who have training requirements defined by laws and regulations (as defined in Section 11.3.8)

- **Operations and Maintenance Personnel Training** for those persons relied upon to operate or maintain IROFS (Section 11.3.10)

- **System Engineer Training** for those persons who review design modifications to IROFS (Section 11.3.11)

The Training Manager is responsible for establishing procedures governing the development and implementation of training programs. The Performance Based Training (PBT) methodology is used for those tasks associated with the design, modification, operation, or maintenance of IROFS.

The Lead Cascade functional organization managers are responsible for defining the job-specific training needs and ensuring completion of training and qualification for personnel within their organization. Workers relied upon to design, operate, or maintain IROFS are trained and evaluated for qualifications prior to assignment of these duties. Task or duty area qualification is granted by line management based on successful evaluation of the worker's mastery of the learning objectives presented during training. Maintenance of qualification is

contingent upon successful completion of continuing training and/or through satisfactory on-the-job training (OJT) evaluations.

Construction personnel, Lead Cascade Operators, technicians, maintenance personnel, and other staff whose actions are relied upon for safety complete the applicable training programs or have equivalent experience or training. Lead Cascade functional organization managers develop and maintain a description of each organization's training requirements. These requirements are identified in Training Requirement Matrices (TRMs) approved by the line and training management. Based on these requirements, the training attendance is tracked by the Training Organization and line management. The Training Organization notifies line management of personnel who have not successfully completed initial training or who are past due for identified continuing training. Line management is responsible for placing work restrictions or removing employees from duty where training is deficient.

11.3.1.1 Initial Training

Initial training contains the classroom and OJT necessary to provide an understanding of the fundamentals, basic principles, systems, procedures, and emergency responses involved in an employee's work assignments. Initial task or duty area qualification is granted by line management based on successful evaluation of the employee's mastery of the learning objectives presented during the training.

Corrective actions involving training are assigned, scheduled and tracked to completion. Lessons learned, which have an impact on initial training, are factored into training materials prior to the delivery of the next training session.

11.3.1.2 Continuing Training

Continuing training is provided for employees in the interest of promoting safety, safeguards and security, and environmental protection awareness. Continuing training is also provided as a means to maintain and improve job-related knowledge and skills.

Plant procedures and TRMs contain training requirements, which delineate continuing training for employees. For those positions that utilize PBT, the training requirements are defined in each Training Development and Administrative Guide (TDAG). The number of hours dedicated to this training annually is determined based on the following factors:

- Frequency required by regulatory agencies and national standards
- Overtrain tasks identified in PBT-based programs
- Training needs as determined by line management. This includes, but is not limited to, nuclear criticality safety assessments, facility or system changes, component changes, procedure changes, lessons learned (including industry and in-house operating experiences, and event reports), and emergency response procedures

11.3.2 Training Participation and Selection

Personnel are selected for entry into the training programs in conformance with the established general employment policies. Operations Technicians, maintenance personnel, and technicians have as a minimum a high school diploma or satisfactory completion of the General Education Development test and three years of industrial/chemical/nuclear plant operations, maintenance, engineering, or support experience. Technician candidates not meeting the experience requirements are placed into entry-level associate technician positions. The minimum education, experience, and qualification requirements for managers and supervisors are described in Chapter 2.0, Organization and Administration, of this license application.

Personnel may be exempted from training as defined in training procedures. New hires or position incumbents may be considered for exemption from segments of classroom training and OJT. Exemptions are based on one of the following methods:

- Management review of an individual's prior training records and/or job performance history provides information demonstrating that the individual has achieved the necessary required skills; or
- Employee demonstrates minimum knowledge requirements by passing module examination in lieu of training (test-out); or
- Employee demonstrates minimum skills/proficiency requirements by successfully completing task performance evaluations in lieu of OJT.

Exemptions are granted when justified, documented, and approved by line and training management.

11.3.3 Performance Based Training

The training program for those personnel who design, operate, or maintain IROFS is based on a PBT methodology. This includes the following elements:

- Conduct of needs/job analysis and identification of tasks for training
- Development of learning objectives
- Development of lesson plans and training guides
- Evaluation of trainee mastery of learning objectives
- Evaluation of the effectiveness of training

11.3.3.1 Conduct of Needs/Job Analysis and Identification of Tasks for Training

A needs/job analysis is 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. The analysis is conducted utilizing either written surveys or the tabletop method with subject matter experts. The training programs for the following Lead Cascade job positions/worker classifications are based on a needs/job analysis:

- Operations Technician
- Operations Shift Supervisor
- Maintenance Technician
- Maintenance Support Technician
- Maintenance Shift Supervisor
- System Engineer

The facility-specific task list is developed for each of the above positions/classifications. The task lists are analyzed based on input from line management and subject matter experts, rating each task on degree of difficulty, importance of the task, and frequency of task performance. From this analysis, the tasks are selected for training based on their rating. The ratings are:

- **Overtrain** - requires initial and continuing training;
- **Train** - requires initial training;
- **Pre-train** or **just-in-time** - requires training but is not taught until that specific knowledge or skill is needed; or
- **No train** -formal training is not required.

The tasks selected for training are matrixed to the associated procedures and training materials. The Task to Train to Procedure Matrices are reviewed and updated in conjunction with the periodic review of the associated procedures.

Training materials are linked to the Configuration Management (CM) system to provide reasonable assurance that design changes and modifications are accounted for in the training. The training materials are matrixed to procedures such that design changes or plant modifications are analyzed by line and training personnel for impact on training.

Procedure changes, equipment changes, job scope changes, facility modifications and other changes affecting task performance are monitored and evaluated for their impact on the development or modification of initial and continuing training programs. The affected training materials are modified or new materials developed, based on the significance of the change, and modifications are documented in the program files. The training materials are updated prior to conducting training.

11.3.3.2 Development of Learning Objectives

Learning objectives are established to identify the training content and to define satisfactory trainee performance for the task or group of tasks selected for training. 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 required of the trainee are also determined in development of the learning objectives. Learning objectives are sequenced within training materials based on their relationship to one another.

Learning objectives are documented in lesson plans and training guides and are revised as necessary based on changes in procedures, facility systems/equipment, or job scope.

11.3.3.3 Development of Lesson Plans and Training Guides

Learning objectives derived from the rated task lists are analyzed to determine the appropriate training setting. Classroom lesson plans, OJT guides, or other instructional materials are procured or developed based on this instructional analysis and design. Lesson plans and other training guides provide the guidance and structure necessary to ensure consistent delivery of training material from trainer to trainer and class to class.

Classroom lessons are used primarily to provide cognitive learning on the fundamentals, theory, basic operating and maintenance principles, individual systems, system inter-relations, safety requirements, and processes used in the Lead Cascade.

Other forms of instructional materials, such as video, computer-based training and self-study may be used as alternatives or supplements to classroom instruction.

OJT is a systematic method of providing training on job-related skills and knowledge for a position. This training is conducted in the work environment and demonstrates actual task performance whenever practical. Applicable tasks and related procedures for each technical area provide the input for the OJT that is designed to supplement and complement training received through formal classroom or laboratory training and to ensure personnel are qualified to perform their assigned tasks.

Classroom lesson plans, OJT guides, and other instructional materials receive technical reviews by designated subject matter experts (SMEs) and instructional reviews by training management as part of the approval process. The responsible line and training managers approve training materials before issuance.

Designated SMEs or technical trainers provide classroom training and/or OJT evaluations. These personnel receive training and are qualified on the instructional methods and techniques applicable to the training setting.

TDAGs describe programs developed for a specific PBT-based job/position. This description includes:

- Organization and Administration Responsibilities

- Trainee Selection Criteria
- Course Loading for Initial and Continuing Training
- Training Resource and Facility Guidelines
- Test/Evaluation Guidelines
- Training and Evaluation Documentation Guidelines
- Course/Modules for Specific Qualification Areas (listed by title and numbers)

11.3.3.4 Evaluation of Trainee Mastery of Learning Objectives

Within the job position/worker classification training programs are logical instructional blocks or "modules" presented in such a manner that specific learning objectives are accomplished. Trainee progress is evaluated by line and training management through a variety of performance demonstrations such as written examinations, oral examinations, and practical tests to ensure mastery of the job performance requirements or learning objectives contained in these modules. Remediation is provided as appropriate.

11.3.3.5 Evaluation of the Effectiveness of Training

Systematic evaluations of training effectiveness and its relation to on-the-job performance are used to ensure that the training program conveys required skills and knowledge and to revise the training, where necessary, based on the performance of trained personnel in the job setting. The student feedback of the training received and the line manager's evaluation of the student's performance on the job after training is completed are utilized to determine the training effectiveness and areas for refinement. Student feedback occurs at several points in the training program. At the completion of training, the student evaluates the instructor and course. Post training evaluations of the effectiveness of training is requested from students and supervisors after completion of training. Each of these evaluations is specified in facility training procedures.

Line and training management conduct self-assessments and evaluations of the individual training programs. Quality Assurance (QA) auditors provide additional assessments through the audit program. These assessments and evaluations are used to determine training program strengths and weaknesses.

11.3.3.6 Training Instructor/Developer Qualification

Training Instructor/Developer Qualification is the responsibility of the Training Organization. Training is provided to designated SMEs and/or technical trainers who develop and/or conduct classroom training and/or OJT evaluations for Lead Cascade personnel. The program consists of modules designed to train instructor/developers in the application of the PBT methodology.

Instructors and subcontractors hired to develop training materials have ready access to designated subject matter experts who assist them when developing training materials. All training program materials are reviewed and approved by subject matter experts and line management prior to implementation.

11.3.4 General Employee Training

GET provides awareness level training on the hazards and proper response to alarms that a person may encounter at the PORTS reservation. It is required biennially for all personnel having unescorted access to the reservation. GET includes the following subject areas:

- General Employee Radiological Safety
- Nuclear Criticality Safety
- General Topics
- Hazard Communication
- Emergency Preparedness

11.3.4.1 General Employee Radiological Safety

General Employee Radiological Training covers the individual's responsibilities for maintaining exposures to radiation and radioactive materials in accordance with the ALARA philosophy. This training reviews natural background and manmade sources of radiation, the whole body radiation dose limit for non-radiological workers, the potential biological effects from chronic radiation doses, embryo and fetus protection, ALARA concepts and practices, and methods used to control radiological materials and contamination. If a person requires unescorted access to a radiological restricted area, additional radiological safety training is provided as discussed in Section 11.3.6 of this license application.

11.3.4.2 Nuclear Criticality Safety

An overview of the Nuclear Criticality Safety (NCS) program is provided. The training emphasizes the prevention of accidental nuclear criticality, describes the hazards and risks of a nuclear criticality accident, explains NCS responsibilities, and teaches the proper response to a nuclear criticality alarm.

Additional NCS training based on American National Standards Institute (ANSI)/American Nuclear Society (ANS) ANSI/ANS-8.20-1991, *American National Standard for Nuclear Criticality Safety Training*, is provided for personnel who handle or manage the handling of fissile material and work within Fissile Material Operations Areas.

11.3.4.3 General Topics

General Topics include a general overview of: (1) health and safety awareness programs; (2) the employee's rights and responsibilities and the employer's duties as defined by laws and regulations; and (3) use of procedures and conduct of operations.

11.3.4.4 Hazard Communication

The purpose of this awareness-level training is to inform personnel that hazardous chemicals are present in the work place and to help them understand the function of warning labels and signs, Material Safety Data Sheets, and the written Hazard Communication Program.

Additional chemical safety training is provided to those personnel who handle or supervise the handling of hazardous chemicals identified in Chapter 6.0, Chemical Process Safety, of this license application.

11.3.4.5 Emergency Preparedness

This training introduces personnel to basic Emergency Plan elements including: (1) emergency plan safety objectives and priorities; (2) ways to report emergencies; (3) recognition and correct responses to facility alarm signals; (4) evacuation guidelines for radiological and non-radiological emergencies; (5) personnel accountability procedures; (6) fire extinguisher familiarization; and (7) personnel responsibilities during emergencies.

11.3.5 Visitor Site Access Orientation

Site access training is provided for plant visitors who are escorted. It utilizes self-study of an orientation handbook and covers the following general information:

- Driving Rules
- Compliance with postings and signs
- Use of eye, head, hearing, and respiratory protection
- Emergency Phone Numbers
- Radiological protection concerns
- Emergency Preparedness
- Security requirements and limitation of access and items prohibited

11.3.6 Radiation Worker Training

Radiation Worker Training is a biennial training requirement for personnel whose job requires them to have unescorted access to radiological restricted areas. The training includes a comprehensive curriculum consisting of the following, as appropriate:

- Fundamentals of atomic structure, radiological definitions, types of ionizing radiation, units of measurement, dose, and dose rate calculations

- Biological effects of ionizing radiation including cell sensitivity and chronic and acute exposure

- Radiation work permit applications and use

- Radiation limits for occupational and non occupational workers as well as the general public

- ALARA practices for protection from exposure to radiation or radioactive materials

- Personnel Monitoring Programs in place to monitor the worker's exposure to radiation

- Radioactive Contamination Control to minimize and control the spread of contamination

- Radiological Postings and Controls for familiarization with the signs and postings in the work area

- Emergencies involving radiological material and the correct response

- Chemical Toxicity of Soluble Uranium Compounds

This training includes knowledge examinations and practical factor examinations of the personal protective equipment, personnel monitoring, and radiation measurements, if needed. Radiation Worker Training is reviewed and approved by the Radiation Protection Manager and administered by the Training Organization. The extent of the course material is commensurate with the potential for exposure. The Radiation Worker Training program is reviewed and evaluated every two years.

11.3.7 Nuclear Criticality Safety Training

NCS training based on ANSI/ANS-8.20-1991 is provided for personnel who handle or manage the handling of fissile material and work within Fissile Material Operations Areas. This

training is reviewed and approved by the NCS technical staff and includes a discussion of the following:

- The fission process
- Controllable factors and examples of their application at this facility
- NCS postings
- NCS emergency procedures
- Consequences of historical criticality accidents

Personnel are trained to report defective or off-normal NCS conditions to the responsible manager of the operation or the Operations and Maintenance Managers and to perform actions only in accordance with written, approved procedures. Personnel are trained that unless a specific procedure deals with the situation, they shall take no action until the NCS personnel have evaluated the situation and provided recovery guidance. NCS refresher training is required every two years.

Managers of personnel described above receive additional training on the managerial responsibilities relating to NCS.

11.3.8 Environmental, Safety, and Health Training

This training covers those environmental, worker safety, and health subject areas required by applicable local, state and federal regulations and is provided to personnel commensurate with their job assignments. Specific modules identified as required compliance training for Lead Cascade employees is contained in each individual's training requirement matrix. Some of the areas include:

- Radiological Worker Safety
- Nuclear Criticality Safety
- Respiratory Training
- Hearing Conservation
- Occupational Safety and Health Administration (OSHA) Hazard Communication
- Hoisting and Rigging
- Mobile Equipment Operations
- Lockout/Tagout Work Permits

- Safety and Health Work Permits
- *Resource Conservation and Recovery Act* for Hazardous Waste Generators
- OSHA Hazardous Waste Operations and Emergency Response Standard
- Personal Safety

11.3.9 Security Education

Security Education briefings are described in the Security Program. The briefings are described as Initial Briefings, Refresher Briefings, Termination Debriefings, and Foreign Travel Briefings.

11.3.10 Operations and Maintenance Technical Training

Training is designed, developed, and implemented to assist facility employees in gaining an understanding of applicable fundamentals, procedures, and practices specific to the Lead Cascade. It is also used to develop the skills necessary to perform assigned work in a safe manner. If a task is identified to operate or maintain an IROFS, then the PBT methodology is used. Initial and continuing training is provided for the following operations and maintenance job categories relied on to operate and/or maintain IROFS.

- **Operations Technician** - This program is designed for personnel who monitor and operate centrifuge equipment and systems. They operate all systems within the Lead Cascade necessary to support the Lead Cascade, perform integrated system testing, execute valving orders, adjust equipment settings, start-up, and shutdown equipment. Operations Technicians also assemble, transfer, install, repair and test centrifuge machines. The Operations Technician training and qualification program is separated into three sequential phases:

  Phase I provides classroom training on basic fundamentals and consists of the following: Centrifuge Operations Orientation, Uranium Enrichment Technology, Operating Principles and Theory of Centrifuge Equipment, Process Control, and Process Support Systems.

  Phase II provides classroom and OJT on the design, assembly, transport, and repair of centrifuge machines.

  Phase III provides classroom and OJT on the ISA; NCS limits and controls; equipment operations; support systems; and normal, off-normal, and emergency operating procedures for the Lead Cascade.

- **Operations Shift Supervisor** - This program is designed for personnel who supervise the Operations Technicians and make operational decisions during normal, off normal, and emergency operations. The Operations Shift Supervisor is the

senior person on shift and directs equipment start-up and shutdown and changes in system alignments. The Operations Shift Supervisor training and qualification program is separated into four sequential phases:

Phase I provides classroom training on basic fundamentals and consists of the following: Centrifuge Operations Orientation, Uranium Enrichment Technology, Operating Principles and Theory of Centrifuge Equipment, Process Control, and Process Support Systems.

Phase II provides classroom and OJT on the design, assembly, transport, and repair of centrifuge machines.

Phase III provides classroom and OJT on the ISA; NCS limits and controls; operations; support systems; and normal, off-normal, and emergency operating procedures for the Lead Cascade.

Phase IV provides classroom and OJT on the supervisory roles and responsibilities for the safe operation of the Lead Cascade.

- **Maintenance Support Technician** - This program is designed for maintenance personnel who service and repair computers, programmable controllers, and electrical, electronic, and pneumatic support systems and components. The Maintenance Support Technician training and qualification program is separated into three sequential phases:

  Phase I provides classroom training on Centrifuge Operations Orientation and Operating Principles and Theory of Centrifuge Equipment.

  Phase II provides classroom and OJT on the Lead Cascade electrical, instrument, and electronic control systems and components.

  Phase III provides classroom and OJT on maintenance procedures, programs and practices.

- **Lead Cascade Maintenance Technician** - This program is designed for maintenance personnel who install, remove, repair, and service mechanical equipment and systems in the field and in shop locations. The Lead Cascade Maintenance Technician training and qualification program is separated into three sequential phases:

  Phase I provides classroom training on Centrifuge Operations Orientation and Operating Principles and Theory of Centrifuge Equipment.

  Phase II provides classroom and OJT on the Lead Cascade mechanical systems and components.

  Phase III provides classroom and OJT on maintenance procedures, programs and practices.

- **Maintenance Shift Supervisor** - This program is designed for the supervisors of the Maintenance Technician and Maintenance Support Technicians. The Maintenance Shift Supervisor training and qualification program is separated into four sequential phases:

  Phase I provides classroom training on Centrifuge Operations Orientation and Operating Principles and Theory of Centrifuge Equipment.

  Phase II provides classroom and OJT on the Lead Cascade mechanical, electrical, instrument, and electronic control systems and components.

  Phase III provides classroom and OJT on maintenance procedures, programs, and practices.

  Phase IV provides classroom and OJT on the supervisory roles and responsibilities for the safe operation of the Lead Cascade.

11.3.11 System Engineer Training

System Engineer training is provided to those persons who provide engineering support; review process equipment operational parameters, analyze the data and determine equipment settings; and review of the design and modifications of IROFS. System Engineers are responsible for reviewing design proposals and modifications; ensuring that the appropriate documents and procedures are updated to be consistent with modifications; and assisting in work control preparation and identification of post-maintenance test requirements for IROFS. The System Engineer has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and three years of nuclear experience. The training is based on a review of job analysis data, training requirements for specific systems, and existing training materials.

11.3.12 Nuclear Criticality Safety Engineer/Specialist Training

PORTS NCS personnel administer Nuclear Criticality Analyst training and qualification. Training is based on ANSI/ANS-8.20-1991 and ANSI/ANS-8.19-1996, *Administrative Practices for Nuclear Criticality Safety*. NCS procedures define educational and experience prerequisites, along with required training courses and OJT activities to be completed prior to qualification.

11.3.13 Fire Protection and Emergency Management Training

11.3.13.1 Fire Protection Training

Emergency medical response personnel meet requirements for state certification as emergency medical technician (these 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. State certification requirements provide the basis for firefighter training programs. Drills are conducted quarterly, as part of the PORTS Emergency Plan.

11.3.13.2 Emergency Management Training

Training is conducted in the areas of:

- General Emergency Plan training
- Specialized Emergency Plan training for the Emergency Response Organization
- Offsite Emergency Management training

Emergency Management drills and exercises are conducted to develop, maintain, and test the response capabilities of personnel, facilities, equipment, and training.

11.3.13.3 Plant Shift Superintendent Training

PSS training and qualification is administered in accordance with guidelines provided in the TDAG for the PSS. It utilizes the performance based training methodology. This training provides the PSS an understanding of the overall integration of the processes, support systems, administrative and emergency procedures, and regulatory reporting requirements necessary to support the Lead Cascade emergency response needs. The PSS's training program is structured into several distinct, but interrelated courses that include the following:

- Incident Command and Emergency Response
- Occurrence Reporting, Problem Identification, Evaluation, Disposition, and Regulatory Notifications
- Technical Subjects

11.3.14 Health Physics Technician

Health Physics support training and qualification is administered in accordance with guidelines provided in the TDAG for Health Physics Technicians. It utilizes the performance based training methodology and applies to those individuals, both plant and contractor, who are engaged in the evaluation of radiological conditions in the Lead Cascade facilities and the implementation of the necessary radiological safety measures as they apply to nuclear facility workers and members of the general public.

11.3.15 Laboratory Technician

Laboratory support training and qualification is administered by in accordance with the guidelines set down in the Administrative Guide for the Laboratory and Technician Training Program. The training utilizes the performance based training methodology. Training is provided in the areas of Laboratory Controls and Standards, Mass Spectrometry, Process Services, Chemical Technology, Uranium Sampling, and Uranium Analysis.

11.3.16 Maintenance of Training Records

Training attendance records, examinations, employee qualification records, and program needs are maintained in an accurate, auditable manner to document each employee's training. The individual employee training records are maintained by the Training Organization.

11.4 Procedures

A management controls program has been established for the development, issuance, and control of Lead Cascade procedures. Procedures that are not related to nuclear safety, safeguards, and security and do not involve or impact the facility, facility operation as described in the ISA Summary, or the assumptions or conclusions of the ISA Summary are not governed by the requirements of this section.

11.4.1 Scope

The Licensee is committed to the use of approved and controlled written procedures to conduct nuclear safety, safeguards, and security activities for the protection of the public, facility employees, and the environment. Procedures are used to ensure safe work practices and apply to workers, visitors, contractors, and vendors. Activities involving nuclear material and/or IROFS are conducted in accordance with approved procedures.

A balanced combination of written guidance, craftsman skills, and work site supervision is utilized. The procedure process utilizes a graded approach to provide the necessary rigor for safe Lead Cascade operation, assure the Licensee's commitments to meeting regulations and standards, and assure a balance of effective safety with practical efficiency in facility operations.

Procedures are intended to prescribe those essential actions or steps needed to safely and consistently perform operations and maintenance activities. These elements are outlined in a procedures management writer's guide and described in implementing procedures.

11.4.2 Procedure Hierarchy

The procedure hierarchy is established in four levels. The levels are:

- Level 1 - Policy statements issued by executive management that apply to Lead Cascade personnel
- Level 2 - Standard Practice Procedures that apply to more than one organization
- Level 3 - Procedures issued at the organization level that apply to more than one group within a larger group or specific organization
- Level 4 - Procedures issued within a group or sub-function

11.4.3 Procedure Types

The following types of procedures are used by the Licensee in the Lead Cascade:

- Administrative Procedures: Procedures that deal with policy or programs/administrative systems, provide programmatic requirements, and do not normally involve manipulation of equipment

- Operating (non-administrative) Procedures: Procedures that direct or cause operation/maintenance of equipment or may directly affect any physical characteristics of equipment

- Alarm Response Procedures: Procedures that provide information that identifies the symptoms of the alarm, possible causes, automatic actions, the immediate operator action to be taken, and the required supplementary actions

- Off-Normal Procedures: Procedures that describe actions to be taken during unusual or out-of-the ordinary situations

- Emergency Operating Procedures: Procedures directing actions necessary to mitigate potential events or events in progress that involve needed protection of on-site personnel, the public health and safety, and the environment

11.4.4 Procedure Process

Procedures are developed or modified through a formal process incorporating the change controls described in Section 11.1 of this license application. The procedure process utilizes nine basic elements to accomplish procedure development, review, approval, and control: Identification, Development, Review and Comment Resolution, Verification, Validation, Approval, Issuance, Change Control, and Periodic Review. These elements are discussed in the following sections.

11.4.4.1 Identification

As a minimum, a procedure is required for:

- The operation of IROFS and the management measures supporting those IROFS as identified in the ISA Summary

- Operator actions necessary to prevent or mitigate the consequences of accidents described in the ISA Summary

- Safe work practices to control processes and operations with special nuclear material, IROFS, and/or hazardous chemicals incident to the processing of licensed material

Maintenance activities can be addressed by written procedures, documented work instructions, or drawings appropriate to the circumstances as discussed in Appendix A.6, paragraph (a), of ANSI/ANS 3.2-1994, *Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants*.

Any new or revised NRC requirements that are promulgated are evaluated to determine the impact on existing implementing procedures or to identify the need for new implementing procedures. Procedures are reviewed following unusual incidents to determine if changes are appropriate based on the root cause and corrective action determination for the particular incident. Procedure changes that are necessary as a result of a system modification are addressed in Section 11.1 of this license application, as part of the modification control process.

Lead Cascade organization managers have the responsibility for identifying which tasks will be proceduralized within their areas of control, using the criteria in the preceding paragraphs and Section 11.4.8 of this license application.

A procedure is normally not needed if:

- The work is not complex or only involves a few actions (unless failure to properly conduct those actions could result in significant consequences)

- The task requires those skills normally possessed by a qualified person (otherwise known as “skill-of-the-craft”)

- The consequences of an error would be minimal

11.4.4.2 Development

Procedure development and quality is the user organization's responsibility. Procedure development is accomplished in accordance with procedural guidance. A general description follows:

- A system is in place to track and document the procedure process.

- Interviews with procedure users and process walkdowns are utilized to ensure procedures are usable; reflect as-built conditions and process operations; and maintain management controls for nuclear safety, safeguards, and security.

- The procedure use category is determined. This determination documents the designation of a procedure as In-Hand (Continuous Use), Reference Use, or Information Use. The designation is based on the administrative or non-administrative use of the procedure, and the safety or financial consequences of failing to adhere to procedural requirements. Procedure use is discussed in Section 11.4.5 of this license application.

- As the procedure is drafted, attributes that enhance procedural use are included, such as standard style organization, format, cautions, and warnings. Safety limits and IROFS will be clearly identified as such in the procedure.

- Operating procedures are used to directly control process operations at the workstation and include, as necessary, direction for normal operations, off-normal operations, maintenance, alarm response, and emergency operations caused by failure of an IROFS or human error. These procedures provide reasonable assurance of NCS, chemical safety, fire safety, emergency planning, and environmental protection.

- Operating procedures contain the following elements, as applicable:
  - Purpose of the activity
  - Regulations, policies, and guidelines governing the procedure
  - Type of procedure
  - Hazards and safety considerations
  - Operating limits
  - Precautions necessary to prevent exposure to hazardous chemicals (resulting from operations with special nuclear material) or to licensed special nuclear material
  - Hold points or safety checkpoints are identified at appropriate steps

- Measures to be taken if contact or exposure occurs
- IROFS associated with the process and their functions
- The timeframe for which the procedure is valid
- Steps for each operating process phase
- Initial start-up
- Normal operations
- Off-normal operations
- Emergency operations
- Temporary operations
- Normal shutdown
- Emergency shutdown
- Start-up following an emergency or extended downtime

• Maintenance procedures involving IROFS for corrective and preventative maintenance, functional testing after maintenance, and surveillance maintenance activities describe:

  - Qualifications of personnel authorized to perform the maintenance or surveillance
  - Controls on and specification of any replacement components or materials to be used
  - Post-maintenance testing to verify operability of the equipment
  - Tracking and records management of maintenance activities
  - Safe work practices (e.g., lockout/tagout, confined space entry, moderation control or exclusion area, radiation or hot work permits, and criticality, fire, chemical, and environmental issues)
  - Pre-maintenance activities require reviews of the work to be performed, including procedure reviews for accuracy and completeness

  - Steps that require notification of all affected parties (operators and supervisors) before performing work and on completion of maintenance work. The discussion includes potential degradation of IROFS during the planned maintenance

- Input and review by affected parties is required. Other selected reviews are obtained, such as QA to ensure that QA requirements are identified and included in operating procedures.

- The approval process for the procedure is described in Section 11.4.4.5 of this license application.

11.4.4.3 Review

Drafts of new procedures and procedure changes are distributed for technical reviews, safety discipline reviews (e.g., nuclear criticality, fire, radiation, industrial, and chemical process safety), and cross-discipline reviews, as needed.

Functional area and cross-discipline reviews are performed for the new procedure or procedure change. Comments/questions generated during the review process are resolved with the originating organizations. 10 CFR 70.72 and intent/non-intent screenings are performed for new and changed procedures (except minor administrative changes that are processed according to the procedure process).

In addition, the FSRC will review:

- Each new procedure required by Section 11.4.4.1 of this license application

- Each proposed change to procedures required by Section 11.4.4.1 of this license application if the proposed change constitutes an intent change (i.e., a change in scope, method, or acceptance criteria that has safety significance)

11.4.4.4 Verification/Validation

Verification is a process that ensures the technical accuracy of the procedure and that it can be performed as written. Procedures are verified by the procedure owner/user during the procedure development/change process. There are two basic attributes of the verification process. The first attribute relates to the technical accuracy of the procedure. It ensures that all technical information including formulas, set points, and acceptance criteria are correctly identified in the procedure. The second attribute is administrative, in that it verifies the procedure format and style and that it is consistent with the procedure writing guide. A standard checklist is used to ensure required attributes are included.

The purpose of procedure validation is to ensure that no technical errors or human factor issues were inadvertently introduced during the procedure review process. Validation is required

for new procedures or for intent changes to the procedure. Validation is performed in the field by qualified personnel, and may be accomplished by detailed scrutiny of the procedure as part of a walk-through exercise or as part of a walk-through drill (particularly for emergency or off-normal procedures). If the particular system or process is not available for a walk-through validation, talk-through may be performed in the particular shop or training environment. Performance of procedure validation is documented.

11.4.4.5 Approval

Following the resolution of review comments, procedures are approved. Approval authority rests with the responsible manager.

Managers ensure that appropriate training is completed on new and revised procedures (see Section 11.3 of this license application).

11.4.4.6 Issuance and Distribution

Procedures are issued and controlled in accordance with the RMDC program procedures. Copies of current approved procedures are available to users via electronic and/or hard copy distribution in the work areas.

11.4.4.7 Temporary Changes

Temporary changes to procedures required by Section 11.4.4.1 of this license application can be made, provided:

- The temporary change does not result in a change to the ISA as determined by the 10 CFR 70.72 review

- The temporary change does not constitute an intent change (i.e., a change in scope, method or acceptance criteria that has safety significance)

- The change is documented utilizing the procedure process

These temporary changes to procedures may be used for a period of time, which should not exceed 30 days or a period for which the temporary condition exits whichever is greater. Temporary changes that need to exceed this period are assessed to ensure it is appropriate to extend the use of the temporary change or to process a permanent change.

Temporary changes to procedures may be made permanent once the change is reviewed and approved as required by Section 11.4.4 of this license application.

11.4.4.8 Periodic Review

Approved procedures are periodically reviewed to ensure their continued accuracy and usefulness. Procedures are periodically reviewed according to established criteria. The periodicity of these reviews is based on procedure content as follows:

Periodic Review Cycle	Procedures to Be Reviewed
1 year	Emergency Operating, Alarm Response and procedures dealing with highly hazardous chemicals as defined by the chemical safety program
5 years	Procedures not included as part of the one-year review cycle

When conducting the periodic review, the procedure owner or subject matter expert performs a complete administrative and technical (requirements and references) review ensuring information is complete and accurate and that the procedure is usable as written.

11.4.5 Use and Control of Procedures

In-Hand (Continuous Use) procedures are followed step-by-step and are present in the work area while the task is being performed. In-Hand procedures, approved equipment alignment check sheets (e.g., valve lineups or electrical switching orders), or approved operator aids are developed for IROFS that have:

- Extensive or complex tasks;
- Tasks which are infrequently performed; or
- Tasks in which operations must be performed in a specified sequence.

Reference Use procedures are provided for routine procedural actions that are frequently repeated or of minimal complexity, and can be performed from memory. Reference Use procedures are not required to be present in the work area.

Information Use procedures are followed to implement administrative or programmatic requirements.

Hard copy controlled copies of procedures are marked “Controlled Copy.” Working copies of procedures are marked “Working Copy,” and verified as the latest version prior to use. Information Only copies of In-Hand (Continuous Use) or Reference Use procedures are marked “Information Only” to indicate they are not controlled copies and are not used to perform work.

Procedures may be accessed and used directly from the electronic document management system.

Work is stopped, the system is immediately placed in a safe condition, and corrective actions are initiated if a step of a procedure cannot be performed as written, in accordance with facility procedures.

Responsible managers ensure personnel are trained on the use of procedures and are appropriately trained and qualified on the current version of the procedure as described in Section 11.3 of this license application.

11.4.6 Temporary Procedures

Temporary procedures may be issued only when permanent procedures do not exist to:

- Direct operations during testing, maintenance, and modifications
- Provide guidance in unusual situations not within the scope of permanent procedures
- Ensure orderly and uniform operations for short periods when the facility, a system, or component of a system is performing in a manner not covered by existing permanent procedures, or has been modified or extended in such a manner that portions of existing procedures do not apply

These temporary procedures may be used for a period of time, which should not exceed 60 days, or a period for which the temporary condition must exist, whichever is greater. Temporary procedures that need to exceed the 60 days are assessed to ensure it is appropriate to extend the use of the temporary procedure. These temporary procedures are subject to the same level of review and approval as required for permanent procedures.

11.4.7 Records

Records generated during procedure use are identified in the governing procedure and controlled according to the facility RMDC program practices as described in Section 11.7 of this license application.

11.4.8 Topics to be Covered in Procedures

Activities defined by Section 11.4.4.1 of this license application are the minimum activities that are to be covered by written procedures. In addition, any activity described in Section 11.4.4.1 of this license application and listed below is covered by a written procedure (except for the maintenance activities listed below which may be covered by written procedures, documented work instructions, or drawings appropriate to the circumstances). This list is not intended to be all-inclusive, because many other activities carried out during Lead Cascade operations may be covered by procedures not included in this list. Similarly, this listing is not intended to imply that procedures need to be developed with the same titles as those in the list. This listing provides guidance on topics to be covered rather than specific procedures.

- **ADMINISTRATIVE PROCEDURES (Management Control)**
  - Training
  - Internal audits and inspections
  - Incident investigations and reporting
  - Records Management Document Control (RMDC)
  - Configuration Management
  - Changes in facilities and equipment
  - Modification design control
  - Quality assurance
  - Equipment control (lockout/tagout)
  - Shift turnover
  - Work control
  - Management control
  - Procedures management
  - Nuclear criticality safety
  - Fire protection
  - Radiation protection
  - Radioactive waste management
  - Maintenance
  - Environmental protection
  - Chemical process safety
  - Operations
  - IROFS surveillances
  - Calibration control

- Preventive maintenance
- Procurement

OPERATING PROCEDURES

- **SYSTEM PROCEDURES THAT ADDRESS START-UP, OPERATION, AND SHUTDOWN**
  - Electrical power
  - Ventilation
  - Shift routines, shift turnover, and operating practices
  - Sampling
  - UF_6 cylinder handling
  - UF_6 material handling equipment
  - Decontamination operations
  - Uranium recovery
  - Facility utilities (for example: air, nitrogen, cooling water, sanitary water, site water)
  - Temporary changes in operating procedures
- **ABNORMAL OPERATION/ALARM RESPONSE**
  - Loss of cooling
  - Loss of instrument air
  - Loss of electrical power
  - Fires
  - Chemical process releases

- **MAINTENANCE ACTIVITIES THAT ADDRESS SYSTEM REPAIR, CALIBRATION, INSPECTION, AND TESTING**
  - Repairs and preventive repairs of IROFS
  - Calibration of IROFS
  - Functional testing of IROFS
  - High-efficiency particulate air filter maintenance
  - Safety system relief valve replacement
  - Surveillance/monitoring
  - Piping integrity testing
  - Containment device testing
  - Repair of UF_6 valves
  - Testing of cranes
  - UF_6 cylinder inspection and testing
- **EMERGENCY PROCEDURES**
  - Toxic chemical releases (including UF_6)

11.4.9 References

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

11.5 Audits and Assessments

The Lead Cascade organization implements a system of audits and assessments to help ensure that the health, safety, and environmental programs, as described in this license application are adequate and effectively implemented. The system is designed to ensure comprehensive independent oversight of the QA Program at least once every three years (except as noted below). The system is comprised of two distinct levels of activities. These are audits and assessments.

11.5.1 Audits

Audits are conducted by the QA Organization(s) in accordance with written procedures or checklists by qualified auditors. The auditing organizations are independent from operations

of the Lead Cascade. Audits verify the effectiveness of health, safety, and environmental programs and their implementation and determine the effectiveness of the process being assessed. Audits further verify that Lead Cascade operations are being conducted safely in accordance with regulatory requirements and license application commitments, and the ISA.

These audits and their associated frequencies are conducted in accordance with Section 18 of the QAPD and use written procedures or checklists. Audits are performed under the direction of a Lead Auditor, qualified in accordance with the American Society of Mechanical Engineering (ASME) NQA-1, Supplement 2S-3. Lead Auditors and staff auditors are functionally and organizationally independent of the programs and activities that are examined. Where appropriate, audit teams are supplemented with facility and/or external technical specialists.

In addition to periodically evaluating aspects of the QAPD, audits are conducted for the areas of radiation safety, nuclear criticality safety [every two years], chemical safety, fire safety, environmental protection, emergency management, quality assurance, CM, maintenance, training and qualification, procedures, incident investigation, records management, nuclear safety, security, and operations.

Audit results are documented and reported to Lead Cascade senior management as specified in facility procedures. Provisions are made for reporting and corrective action, where warranted. A Lead Cascade Corrective Action Program, described in Section 11.6 of this license application, is administered by the Regulatory Organization to ensure proper control of corrective actions as defined in Section 16 of the QAPD.

11.5.2 Assessments

Management responsible for implementing portions of the QAPD performs assessments to verify the adequacy of the part of the QAPD for which they are responsible and to assure its effective implementation. Results of assessments are documented. The responsible organization manager resolves any observations from these programmatic assessments.

Organization managers maintain an assessment process within their organization to assess the adequacy of, and effectiveness of, the implementation of the programs under their cognizance. As a minimum, these assessments are conducted for the areas of radiation safety, nuclear criticality safety, chemical safety, fire safety, environmental protection, emergency management, quality assurance, CM, maintenance, training and qualification, procedures, incident investigation, records management, nuclear safety, and operations.

Assessment results are documented and reported as specified in Lead Cascade procedures. Provisions are made for reporting and corrective action, where warranted, in accordance with the Lead Cascade Corrective Action Program.

Additional requirements for performing Nuclear Criticality Safety Assessments are specified in Chapter 5 of this license application.

11.6 Incident Investigations

This section encompasses the identification, reporting, and investigation of abnormal events or conditions, including precursor events that may occur during operation of the Lead Cascade. This includes identification and categorization of the incident, as well as an analysis to determine the specific or generic causes, as well as generic implications. The Licensee is required by 10 CFR 70.50 and 70.74 to notify the NRC of certain facility events and conditions and to determine the root cause of the event, including all factors that contributed to the event and the manufacturer and model number (if applicable) of any equipment that failed or malfunctioned. Corrective actions taken or planned to prevent occurrence of similar or identical events in the future and the results of any evaluations or assessments must also be provided.

The Licensee satisfies these requirements by following administrative procedures relating to incident identification and reporting. These procedures work together to ensure that abnormal events and conditions occurring at the facility are promptly reported to appropriate Lead Cascade personnel, assessed, and when required, reported to the NRC Operations Center or designated NRC office.

11.6.1 Incident Identification

In accordance with procedures, plant personnel are required to report to their line manager or directly to the PSS abnormal events or conditions that may have the potential to harm the safety, health, or security of on-site personnel, the general public, or the environment, including precursor events. These conditions may require an emergency response.

11.6.2 Incident Categorization

The PSS, in accordance with procedures, assesses and categorizes abnormal events or conditions using the notification and reporting criteria set forth in 10 CFR 70.50 and 70.74 and other applicable regulations. In making the assessment, the PSS may consult with Lead Cascade management or other personnel possessing expertise or knowledge concerning the type of event or condition being assessed.

11.6.3 Nuclear Regulatory Commission Notification

If an event or condition within the Lead Cascade area is categorized as a reportable event, the PSS makes initial notification to the NRC Operations Center or designated NRC office and provides, to the extent known at the time of notification, the information specified in 10 CFR 70.50(c)(1). Notification is made as soon as possible but not later than the time period stated in the regulations. Notification time periods vary between one and 24 hours. Verbal and/or written communication involving classified information is conducted in accordance with Chapter 2 of the Lead Cascade Security Program.

11.6.4 Conduct of Incident Investigations

The level of investigation of abnormal events and precursor events is based on a graded approach relative to the severity of the incident. Each reportable event where a follow-up written

report to the NRC is required is investigated to determine the root cause and corrective actions necessary to prevent recurrence. This investigation is conducted and documented in accordance with procedures. Other events not requiring a written report are evaluated using the Corrective Action Program to determine actions to be taken.

The investigation process includes a prompt risk-based evaluation and, depending on the complexity and severity of the event, one individual may suffice to conduct the evaluation or an event investigation team may be warranted. Investigations will begin within 48 hours of the abnormal event, or sooner, depending on the safety significance of the event and commensurate with the safety of the investigators. A procedure provides a documented plan for investigating abnormal events. This plan is separate from any required Emergency Plan or emergency response. A reasonable, systematic, structured approach is used to determine the specific or generic root causes and generic implications of abnormal events, such as the TapRooT® methodology. The record of IROFS failures required by 10 CFR 70.62(a)(3) for IROFS is reviewed as part of the investigation and updated in accordance with regulatory requirements.

For each event or condition that requires a follow-up written report to the NRC, the incident investigation report includes a description, contributing factors, a root cause analysis, and findings and recommendations. Auditable records and documentation related to abnormal events, investigations, and root cause analyses are maintained. Documentation relating to the investigation is retained for two years or for the life of the operation, whichever is longer. The original investigation reports are available to the NRC upon request.

11.6.5 Investigator Functions, Qualifications, and Responsibilities

The investigator(s) are independent from the line function involved with the incident under investigation and have the authority to obtain all the information considered necessary during the course of the investigation. Line management cooperates fully with the investigators and participants of an investigation team are assured of no retaliation for participation in an investigation. The individual leading the investigation is trained and qualified in root cause analysis techniques. This individual is responsible for ensuring the conduct of the investigation is in accordance with procedures and that the outcome of the investigation is properly documented and reported to appropriate levels of management with responsibility for the abnormal event. If a team is used, it includes at least one process expert in addition to the trained root cause investigator. An individual is chosen to lead the incident investigation based on experience and knowledge of the particular area involved with the event or condition.

11.6.6 Follow-up Written Report

When required by regulations, a report summarizing the results of the event investigation is prepared in accordance with procedures. The report contains, at a minimum, the information specified in 10 CFR 70.50(c)(2). The written report is forwarded to the NRC within the time limit specified in the applicable NRC regulations, with the exception that the follow-up written reports required by 10 CFR 70.50(c)(2) are submitted within 60 days.

11.6.7 Corrective Actions

For each significant condition adverse to quality or reportable event where a follow-up written report to the NRC is required, corrective actions to prevent recurrence are developed by responsible management, tracked in a database, and monitored through completion in accordance with the corrective action program. Corrective actions are taken within a reasonable period, commensurate with the safety significance of the event. Record revisions necessitated by post-failure investigation conclusions will be made promptly in accordance with 10 CFR 70.62(a)(3) based on the nature of the record, extent of revision necessary, and potential safety significance. All necessary record revisions will be made within 30 days of the completion of the investigation, unless specifically approved by Lead Cascade management. Evidence files used to support action closure are maintained in accordance with approved records management procedures.

11.6.8 Lessons Learned

Documentation is maintained so that "lessons learned" may be applied to future operations of the facility. Details of the event sequence are compared with accident sequences already considered in the ISA. Should it be necessary, the ISA Summary is modified to include evaluation of the risk associated with accidents of the type actually experienced. Relevant findings from incident investigations are reviewed with affected Lead Cascade personnel.

11.7 Records Management and Document Control

RMDC programs are established to ensure records and documents required by the QAPD are appropriately managed and controlled. These programs are designed to meet the specific record keeping and document control requirements set forth in 10 CFR Part 70 and the applicable provisions of other parts of 10 CFR. These programs provide administrative controls that establish standard methods and requirements for collecting, maintaining, and disposing of records. These programs also ensure that documents are controlled and distributed in accordance with identified written requirements and authorizations. The administrative controls for the generation and revision of records and documents are contained in implementing procedures. The principal elements of each of the RMDC programs and a brief description of the manner in which the functions associated with each element are performed are provided below, along with a list of the types of records that are retained for the duration of the licensed activities.

11.7.1 Records Management Program

The Records Management program provides direction for the handling, transmittal, storage, and retrievability of records. Records Management design provides for adequate assurance that the appropriate records of IROFS are maintained in accordance with the BDC contained in CFR 70.64(a) and the defense in depth requirements of 10 CFR 70.64(b). Records maintained pursuant to 10 CFR Part 70 may be the original, a reproduced copy, electronic media, or microform, if such reproduced copy, electronic media, or microform is duly authenticated by authorized personnel and is capable of producing clear, complete, accurate and legible copies through storage for the period specified by regulation. Records such as letters, drawings, and specifications must include all pertinent information such as stamps, initials, and signatures. Initials and signatures may be authenticated electronic reproductions**.** Records are categorized

and handled in accordance with their relative importance to safety and storage needs. Special provisions are made for handling contaminated records and ensuring their inclusion in the program. This program is implemented through procedures that provide guidance for the following program elements.

11.7.1.1 Legibility, Accuracy, and Completeness

Documents designated to become records must 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.

11.7.1.2 Identification of Items and Activities

Records clearly and specifically identify the items or activities to which they apply.

11.7.1.3 Authentication

Records are authenticated or validated by the manager of the organization that originates the record, or his designee, as specified in the procedure, which controls the generation and revision of these records. This is in the form of a signature and date applied to the record.

11.7.1.4 Indexing and Filing

Methods are specified for indexing, filing, and locating records within the record system to ensure the records can be retrieved in a timely manner.

11.7.1.5 Retention and Disposition

Records retention times are specified in a retention schedule, developed by the manager of the organization that originates the record, or the designee. The process for disposition of records that have reached the end of their retention lifetime is specified by procedures and conforms to applicable requirements.

11.7.1.6 Corrections

Corrections to records are approved by the organization that created the record unless other organizations are specifically designated. Changes are made by clearly indicating the correction, the date of the correction and the identification of the individual making the correction.

11.7.1.7 Protection of Records

Controls are established for protection of records from deterioration, loss, damage, theft, tampering, and/or unauthorized access for the life of the record. Requirements include instructions on protection of records by the record originator until they are transferred to Records Management. Instructions for the protection of special record media such as radiographs,

photographs, negatives, microform and magnetic media are provided to prevent damage from excessive light, stacking, electromagnetic fields, temperature, humidity, or any other condition adverse to the preservation of those records. Records, which cannot be duplicated, are stored in a fashion that minimizes deterioration.

11.7.1.8 Storage Requirements

Records encompassed by the QAPD are stored in authorized facilities or containers providing protection from fire hazards, natural disasters, environmental conditions, and infestations of insects, mold, or rodents. Storage facilities are maintained to ensure continuous protection of the records. Requirements are specified for both permanent and temporary storage of records.

- **Permanent Storage**

  Records are permanently stored in facilities satisfying the following requirements:

  - Storage in 2-hour-rated containers meeting National Fire Protection Association (NFPA) 232-1986 or NFPA 232 AM-1986 or both, with the clarification that if the NFPA 232 (or 232 AM) method of storage in 2-hour-rated containers is used, any exceptions to this standard will be documented and justified by the authority having jurisdiction; or

  - Storage of duplicate copies in separate facilities that are sufficiently remote from each other to eliminate the possibility of exposure to simultaneous hazards; or

  - Storage in facilities that have the following: doors, structures, frames, and hardware that comply with a minimum 2-hour fire rating; a fire protection system; 2-hour fire rated dampers on all boundary penetrations; sealed floor surface to minimize concrete dust; adequate access and aisle ways; and a prohibition on eating, drinking, or smoking and performing work other than that associated with records storage or retrieval.

- **Temporary Storage**

  The RMDC process requires that those completed records documenting nuclear safety or safeguards and security matters, that are being held temporarily by originating organizations, be properly protected by maintaining them in 1-hour, fire-rated containers. If 1-hour fire-rated containers are used they either bear an Underwriters Laboratory label (or equivalent) certifying 1-hour fire protection, or the containers are certified for 1-hour fire protection by an authorized individual competent in the field of fire protection. Procedural requirements are used to limit the length of time during which records may be maintained in temporary storage, based on the significance of the record.

11.7.1.9 Receipt of Records

A record transmittal process is used to formally transmit records to Records Management. The process includes a receipt acknowledgment that notifies the sending organization that the records have been received and accepted.

11.7.1.10 Access to Records and Accountability for Removed Records

Requirements for controlling access to records and maintaining accountability for records are provided to ensure that only authorized personnel have access to records and to prevent loss, damage, or inadvertent destruction of records.

11.7.1.11 Records Requirements for Procured Goods or Services

Records management requirements for goods or services procured from outside suppliers are specified in the applicable procurement documents. These requirements cover:

- Supplier methods for collection, storage, and maintenance of records
- Identification of required records and applicable retention periods
- Records submittal plans or indexes
- Availability, accessibility, and if applicable, disposition criteria for records retained by the supplier
- Accessibility of the supplier's records prior to the final transfer to the purchaser

11.7.1.12 Control of Sensitive Records

Control, accountability, protection, and disposition of classified, Unclassified Controlled Nuclear Information (UCNI), and sensitive records are in accordance with Chapter 2 of the Security Program for the American Centrifuge Plant and any other applicable security and privacy requirements. Control of contaminated records is in accordance with applicable radiological control requirements.

11.7.1.13 Types of Records

The requirements for records management vary according to the nature of the facility and the hazards and risks posed by it. Examples of the records required by 10 CFR Parts 19, 20, 21, 25, and 70 are identified in Section 11.7.5. The records are listed under the chapter headings of the Standard Review Plan (SRP). The list is not intended to be exhaustive or prescriptive. Different or additional records may be required in certain circumstances.

11.7.1.14 Usage and Control of Computer Codes and Data

Computer programs used in the Records Management program are controlled and maintained in accordance with classified information systems security administration procedure requirements, unclassified computer security plan requirements, and information technology operations guidance. These requirements and practices provide for virus protection as well as access control to the Records Management program database and ensure continuing usability of the codes as hardware and software technology change. Routine backups of the Records Management database are performed by Corporate Information Technology (CIT) application administrators. Precautions are taken to ensure that computer data that constitute a record are stored in a format that is readily retrievable even as hardware and software technology evolve. The storage format of computer data is reviewed as required to determine threats to future retrievability, and if necessary, the data are translated to an updated format and verified acceptable.

11.7.1.15 Assessment

The overall effectiveness of the Records Management program is evaluated through the audit program described in the Section 18 of the QAPD. Deficiencies identified are corrected in a timely manner in accordance with Section 11.6 of this license application.

11.7.2 Document Control Program

The Document Control program provides direction for the handling, distribution, and transmittal of documents important to nuclear safety and safeguards and security that specify quality requirements or prescribe activities affecting quality, such as procedures, drawings, and calculations. This program is implemented through procedures that provide guidance on the following program elements.

11.7.2.1 Unique Identifier

A unique identification number is assigned or obtained by the generator for each document requiring controlled distribution. Document Control concurs with the numbering scheme for each document type.

11.7.2.2 Approval and Release of Documents

For documents and changes to documents required by the QAPD, requirements are established for approval and release of those documents for distribution. Organizations that are authorized to approve controlled documents are identified in the Lead Cascade procedures. Changes to controlled documents are approved. After approval, the documents are forwarded to Document Control for control and distribution pursuant to the personnel on the approved distribution list.

11.7.2.3 Master Copy

A master copy of all approved controlled documents is maintained by Document Control to ensure the document is available for controlled copy issuance.

11.7.2.4 Controlled Document Index and Distribution Lists

Creation and maintenance of a controlled document index and controlled distribution list(s) for each document or document type are required. The controlled document index is used to maintain a list of controlled documents and to track the current (latest) approved revision levels of those documents. The index is available to users to verify current document revision levels. The controlled document index and the distribution lists are maintained and updated by Document Control.

11.7.2.5 Copies of Controlled Documents

Each controlled copy is stamped, marked or otherwise identified. A method is established in procedures for duplicating and marking controlled documents so that duplicates are distinguishable from the controlled version. Copies of controlled documents that are not marked or otherwise identified in accordance with procedural requirements are considered information only.

11.7.2.6 Distribution

Controlled documents are distributed in accordance with controlled distribution lists to ensure that they are available in a timely manner at locations where work is being performed. Specific time requirements are established for controlled document distribution and receipt acknowledgment. Document Control uses a transmittal form to distribute controlled documents to copy holders. Copy holders sign, date, and return the transmittal form to confirm that they have received the documents. Document Control tracks the issuance and receipt of transmittals.

11.7.2.7 Voided, Canceled, or Superseded Documents

When notified by the generator of a controlled document that the document has been voided, canceled, or superseded, Document Control removes the document from distribution and notifies copy holders of the changed status.

The approved revised document is distributed at the time that the original document is superseded. The Document Control database is updated to identify the latest approved revision of the document. Distribution of revised documents is described in the Document Control Program procedure and using a Transmittal Form distributed by either interoffice mail or hand delivery. The holder of the Controlled Copy is required to acknowledge receipt by returning a signed Transmittal Form to Document Control. Document distribution is completed in accordance with the safety significance of the document being distributed.

11.7.2.8 Marking Sensitive Documents

Proper marking and handling of documents designated as classified, UCNI, or sensitive documents is accomplished in accordance with Chapter 2 of the Security Program for the American Centrifuge Plant and any other applicable security and privacy requirements.

11.7.2.9 Change Documents

Change documents are documents that are used to modify controlled documents. Controls are also applied to the change documents to provide revision approval and distribution controls equivalent to the original document until completion of installation, at which time the original document is revised. Documents showing the current configuration are not changed until the modifications are completed.

11.7.2.10 Revision Identification

The controlled document revision level is clearly identified on the document.

11.7.2.11 Document User Responsibilities

Responsibilities of the end user and copy holders are defined. Responsibilities include requirements for the use of controlled documents and working copies. Copy holders of controlled documents update their controlled documents each time a revision or change is sent out, and promptly return the transmittal form acknowledging receipt.

11.7.2.12 Usage and Control of Computer Codes and Data

Computer programs used in the Document Control program are controlled and maintained in accordance with classified information systems security administration procedure requirements, unclassified computer security plan requirements, and information technology operations guidance. These requirements provide for virus protection as well as access control to the Document Control program database and ensure continuing usability of the codes as hardware and software technology change. Routine backups of the Document Control database are performed by CIT application administrators.

11.7.2.13 Assessment

The overall effectiveness of the Document Control program is evaluated through the audit program described in Section 18 of the QAPD. Deficiencies identified are corrected in a timely manner in accordance with Section 11.6 of this license application.

11.7.2.14 Archiving Documents

The record copy of all revisions of controlled documents is transmitted to Records Management in accordance with the requirements of the Records Management program.

11.7.3 Organization and Administration

PORTS provides RMDC services to the Lead Cascade.

11.7.3.1 Responsibilities

The Training Manager is responsible for the Lead Cascade RMDC program. Responsibility for program compliance rests with the Lead Cascade managers generating records. The Training Manager is responsible for:

- Directing all activities and personnel of the RMDC programs
- Directing the development, implementation, and maintenance of methods and procedures encompassing a records management program
- Directing the development, implementation, and maintenance of methods and procedures encompassing a document control program
- Assuring that all laws, codes, standards, regulations, and company procedures pertaining to record keeping and document control requirements are met
- Select RMDC activities may be contracted from a qualified provider.

11.7.3.2 Training and Qualifications

Appropriately trained and qualified personnel manage the RMDC programs. No specific experience related to the control of documents or management of records is required, although previous technical or RMDC experience is recommended.

11.7.4 Employee Training

General training in RMDC is provided to employees as part of the general topics covered in GET, as described in Section 11.3 of this license application.

11.7.5 Examples of Records by Standard Review Plan Chapter

- **Chapter 1.0 - General Information**
  - Construction records
  - Facility and equipment descriptions and drawings
  - Design criteria, requirements, and bases for IROFS as specified by the facility CM function

  - Records of facility changes and associated integrated safety analyses, as specified by the facility CM function
  - Safety analyses, reports, and assessments
  - Records of site characterization measurements and data
  - Records pertaining to onsite disposal of radioactive or mixed wastes in surface landfills
  - Procurement records, including specifications for IROFS

- **Chapter 2.0 - Organization and Administration**
  - Administrative procedures with safety implications
  - Change control records for nuclear material control and accounting program
  - Organization charts, position descriptions, and qualification records
  - Safety and health compliance records, medical records, personnel exposure records, etc.
  - Quality Assurance records
  - Safety inspections, audits, assessments, and investigations
  - Safety statistics and trends

- **Chapter 3.0 - Integrated Safety Analysis**

- **Chapter 4.0 - Radiation Safety**
  - Bioassay data
  - Exposure records
  - Radiation protection (and contamination control) records
  - Radiation training records
  - Radiation work permits

- **Chapter 5.0 - Nuclear Criticality Safety**

- Nuclear criticality control written procedures and statistics
- Nuclear criticality safety analyses
- Records pertaining to nuclear criticality inspections, audits, investigations, and assessments
- Records pertaining to nuclear criticality incidents, unusual occurrences, or accidents
- Records pertaining to nuclear criticality safety analyses

- **Chapter 6.0 - Chemical Safety**
  - Chemical process safety procedures and plans
  - Records pertaining to chemical process inspections, audits, investigations, and assessments
  - Diagrams, charts, and drawings
  - Records pertaining to chemical process incidents, unusual occurrences, or accidents
  - Chemical process safety reports and analyses
  - Chemical process safety training

- **Chapter 7.0 - Fire Safety**
  - Fire Hazard Analysis
  - Fire prevention measures, including hot-work permits and fire watch records
  - Records pertaining to inspection, maintenance, and testing of fire protection equipment
  - Records pertaining to fire protection training and retraining of response teams
  - Pre-fire emergency plans

- **Chapter 8.0 - Emergency Management**
  - Emergency plan(s) and procedures

- Comments on emergency plan from outside emergency response organizations
- Emergency drill records
- Memoranda of understanding with outside emergency response organizations
- Records of actual events
- Records pertaining to the training and retraining of personnel involved in emergency preparedness functions
- Records pertaining to the inspection and maintenance of emergency response equipment and supplies

- **Chapter 9.0 - Environmental Protection**
  - Environmental release and monitoring records
  - Environmental report and supplements to the environmental report, as applicable
- **Chapter 10.0 - Decommissioning**
  - Decommissioning records
  - Financial assurance documents
  - Decommissioning cost estimates
  - Site characterization data
  - Final survey data
  - Decommissioning procedures
- **Chapter 11.0 - Management Measures**
  - Section 11.1 - Configuration Management
    - Safety analyses, reports, and assessments that support the physical configuration of process designs, and changes to those designs
    - Validation records for computer software used for safety analysis or nuclear material control and accounting

- ISA documents, including process descriptions, plant drawings and specifications, purchase specifications for IROFS
- Approved, current operating procedures and emergency operating procedures

– Section 11.2 - Maintenance

- Record of IROFS failures (required by 10 CFR 70.62)
- PM records, including trending and root cause analysis
- Calibration and testing data for IROFS
- Corrective maintenance records

– Section 11.3 - Training and Qualification

- Personnel training and qualification records
- Procedures

– Section 11.4 - Procedures

- Standard operating procedures
- Functional test procedures

– Section 11.5 - Audits and Assessments

- Audits and assessments of safety and environmental activities

– Section 11.6 - Incident Investigations

- Investigation reports
- Changes recommended by investigation reports, how and when implemented
- Summary of reportable events for the term of the license
- Incident investigation policy

– Section 11.7 - Records Management

- Policy

- ➢ Material storage records
- ➢ Records of receipt, transfer, and disposal of radioactive material

– Section 11.8 - Other Quality Assurance Elements

- ➢ Inspection records
- ➢ Test records
- ➢ Corrective action records

11.8 Other Quality Assurance Elements

The Licensee has developed QA principles as described in Section 1.0 of the QAPD.

11.9 References

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

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

3. ANSI/ANS-8.20-1991, *Nuclear Criticality Safety Training*

4. ANSI/ASME NQA-1-1994, *Quality Assurance Requirements for Nuclear Facility Applications*

5. Gas Centrifuge Quality Assurance Program Description, AET 03-0006, Steven A. Toelle letter to Mr. Martin J. Virgilio, dated February 3, 2003

6. NFPA 232-1986, *Standard for the Protection of Records*

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

Appendix A
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