Enclosure 1 of ACO 21-0027

LA-3605-0001, Lic*ense Application for the American Centrifuge Plant*

LA-3605-0001

License Application

for the American Centrifuge Plant
in Piketon, Ohio

Revision 54

Docket No. 70-7004

May 2021

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Export Controlled Information

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Revision 14 – 10 CFR 1045 review completed by J. Weidner on 03/17/06; Export Controlled Information review completed by D. Hupp on 03/17/06.
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Revision 16 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. Coriell on 08/11/06.
Revision 17 – 10 CFR 1045 and the Export Controlled Information reviews were completed by G. Peed on 08/30/06.
Revision 18 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. Coriell on 09/06/06.
Revision 19 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. Coriell on 06/22/07.
Revision 20 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. Coriell on 10/09/07.
Revision 21 – 10 CFR 1045 and the Export Controlled Information reviews were completed by G. Peed on 01/11/08.
Revision 22 – 10 CFR 1045 and the Export Controlled Information reviews were completed by G. Peed on 01/25/08.
Revision 23 – 10 CFR 1045 and the Export Controlled Information reviews were completed by G. Peed on 03/04/08.
Revision 24 – 10 CFR 1045 and the Export Controlled Information reviews were completed by M. Basham on 06/05/08.
Revision 25 – 10 CFR 1045 and the Export Controlled Information reviews were completed by M. Basham on 09/29/08.
Revision 26 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. S. Lykowski on 11/24/08.
Revision 27 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. S. Lykowski on 1/14/09.
Revision 28 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. S. Lykowski on 1/27/09.
Revision 29 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. S. Lykowski on 10/15/09.
Revision 30 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. S. Lykowski on 1/8/10.
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Revision 34 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. S. Lykowski on 7/23/10.
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Revision 36 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. S. Lykowski on 10/6/10.
Revision 37 – 10 CFR 1045 and the Export Controlled Information reviews were completed by R. S. Lykowski on 11/3/10.
Revision 38 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 2/18/11.
Revision 39 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 3/4/11.
Revision 40 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 4/19/2011.
Revision 41 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 9/2/2011.
Revision 42 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 10/20/2011.
Revision 43 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 8/27/2012.
Revision 44 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 2/7/2013.
Revision 45 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 2/20/2013.
Revision 46 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 8/20/2013.
Revision 47 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 11/21/2013.
Revision 48 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 12/16/2013.
Revision 49 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 2/13/2014.
Revision 50 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 7/28/2015.
Revision 51 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier R.S. Lykowski. Sensitive information reviews completed and approved for public release by R.S. Lykowski on 12/2/2015.
Revision 52 - Reviewed and determined to be UNCLASSIFIED. Derivative Classifier #4769. Sensitive information reviews completed and approved for public release by Reviewer #1014 on 6/17/2016.

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Proposed Change – Classification review completed by Derivative Classifier #4734 on March 9, 2021 and the Controlled Unclassified Information (e.g., ECI) review completed by Reviewer #1014 on March 9, 2021.
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Revision 54 – Classification review completed by Derivative Classifier #4734 on March 31, 2021 and the Controlled Unclassified Information (e.g., ECI) review completed by Reviewer #1014 on March 31, 2021.

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APPENDICIES

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ACRONYMS AND ABBREVIATIONS

ACE	American Centrifuge Enrichment, LLC
ACH	American Centrifuge Holdings, LLC
ACL	Administrative Control Level
ACM	American Centrifuge Manufacturing, LLC
ACO	American Centrifuge Operating
ACP	American Centrifuge Plant
ACR	Area Control Room
ACS	Access Control System
ACT	American Centrifuge Technology, LLC
AEA	Atomic Energy Act
AHJ	Authority Having Jurisdiction
AIHA	American industrial Hygiene Association
ALARA	as low as reasonably achievable
amsl	above mean sea level
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARA	Airborne Radioactivity Area
ARF	airborne release fraction
ASME	American Society of Mechanical Engineers
AST	above ground storage tank
ASTM	American Society for Testing and Materials
BCS	Boundary Control Station
BDC	Baseline Design Criteria
BEQ	Baseline Effluent Quantity
BOP	Balance of Plant
BUSTR	Bureau of Underground Storage Tanks
CA	Contamination Area
CAA	Controlled Access Area
CAAS	Criticality Accident Alarm System
CCZ	Contamination Control Zone
CEDE	Committed Effective Dose Equivalent
CERCLA	Comprehensive Environmental Response, Compensation, and Liabilities Act
CFR	*Code of Federal Regulations*
CM	Configuration Management
CVP	Cylinder Valve Protectors
CW	chilled water
CWA	Clean Water Act
CWIP	Construction Work in Progress
D&D	decontamination and decommissioning
DA	Design Authority
DAC	Derived Air Concentration
DBE	design basis earthquake
DCP	double contingency principle
DFP	Decommissioning Funding Plan

D-G	diesel generator
DID	defense in depth
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DP	Decommissioning Plan
DR	damage ratio
DSA	Decontamination Service Area
DUF6	depleted uranium hexafluoride
ECS	Engineering Consulting Services
EOC	Emergency Operations Center
EPA	Environmental Protection Agency
EPC	Engineering, Procurement, and Construction
EPCRA	Emergency Planning and Community Right to Know Act
ERPG	Emergency Response Planning Guidelines
ER	Environmental Report
EV	evacuation vacuum
FBP	Fluor-BWXT Portsmouth
FCA	Fixed Contamination Area
FHA	Fire Hazards Analysis
FM	Factory Mutual
FNAD	Fixed Nuclear Accident Dosimeters
FNMCP	Fundamental Nuclear Material Control Plan
FPPA	*Farm Protection Policy Act*
FSU	former Soviet Union
FWLA	Fugro, Williams, Lettis and Associates
FHA	Fire Hazards Analysis
FNAD	Fixed Nuclear Accident Dosimeters
FNMCP	Fundamental Nuclear Material Control Plan
GCEP	Gas Centrifuge Enrichment Plant
GDP	gaseous diffusion plant
GET	General Employee Training
GTC	Gas Test Stand Center
HA	Hazard Analysis
HALEU	High Assay Low Enriched Uranium
HAZMAT	hazardous material
HCA	High Contamination Area
HE	Hazard Evaluation
HEPA	high efficiency particulate air
HEU	high enriched uranium
HMTA	Hazardous Materials Transportation Act
HP	Health Physics
HRA	High Radiation Area
HVAC	Heating, Ventilation, and Air Conditioning
IC	initial condition
ICP/MS	Inductively Coupled Plasma/Mass Spectrometry
IDS	Intrusion Detection System

IEEE	Institute of Electrical and Electronics Engineers
IEU	intermediate enriched uranium
IHS	Industrial Hygiene and Safety
IPP	Interconnecting Process Piping
IROFS	items relied on for safety
ISA	Integrated Safety Analysis
ISTP	Integrated Systems Test Plan
LCC	local control center
LEC	Liquid Effluent Collector
LEPC	Local Emergency Planning Commission
LEU	low enriched uranium
LLMW	low level mixed waste
LLRW	low level radioactive waste
LPF	leak path factor
LSDA	Lower Suspension and Drive Assembly
M&TE	measuring and test equipment
MAR	material at risk
MCNP	Monte Carlo n-particle
MCS	Mid-America Conversion Services, LLC
MCW	machine cooling water
MDA	Minimum Detectable Activity
MEI	Maximally Exposed Individual
MM	Modified Mercalli
MSDS/SDS	Material Safety Data Sheet/Safety Data Sheet
NA	not applicable
NAAQS	National Ambient Air Quality Standards
M&TE	measuring and test equipment
NCS	Nuclear Criticality Safety
NCSE	Nuclear Criticality Safety Evaluation
NDA	Nondestructive Assay
NEMA	National Electrical Manufacturers Association
NEPA	National Environmental Protection Act
NESHAP	National Emissions Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NHPA	*National Historic Preservation Act*
NIST	National Institute of Standards and Technology
NMC&A	Nuclear Materials Control and Accountability
NPDES	National Pollutant Discharge Elimination System
NPH	natural phenomena hazard
NRC	U.S. Nuclear Regulatory Commission
NSPS	new source performance standards
NVLAP	National Voluntary Laboratory Accreditation Program
OAC	Ohio Administrative Code
OEPA	Ohio Environmental Protection Agency
OJT	on-the-job training
ORC	Ohio Revised Code

OSHA	Occupational Safety and Health Administration
PA	Public Address
PBT	Performance Based Training
PCF	Plant Control Facility
PFPE	polyfluoropolethers
PGA	peak ground acceleration
PBT	Performance Based Training
PHA	Preliminary Hazard Analysis
PM	preventive maintenance
PMF	Probably Maximum Flood
PMT	post-maintenance testing
PORTS	Portsmouth Gaseous Diffusion Plant
PPE	personal protective equipment
PSD	prevention of significant deterioration
PSM	Process Safety Management
PSP	Protective Shipping Packages
PSRC	Plant Safety Review Committee
PSS	Plant Shift Superintendent
PT	performance testing
PTI	permits-to-install
PV	purge vacuum
QA	Quality Assurance
QAPD	Quality Assurance Program Description
QC	Quantity Control
QL	Quality Level
QRA	Quantitative Risk Analysis
R&D	research and development
R/A	Recycle/Assembly
RA	Radiation Areas
RAM	random access memory
RCRA	*Resource Conservation and Recovery Act* of 1976
RCW	recirculating cooling water
REIRS	Radiation Exposure Information Reporting System
RF	respirable fraction
RG	Regulatory Guide
RHW	recirculating heating water
RM	river mile
RMA	Radioactive Material Area
RMC	Ridge Mast Crane
RMDC	Records Management and Document Control
RMP	Risk Management Program
RP	Radiation Protection
RPM	Radiation Protection Manager
RQ	Reportable Quantity
RWP	Radiation Work Permit
SAR	Safety Analysis Report

SARA	*Superfund Amendments and Reauthorization Act*
SERC	Ohio State Emergency Response Commission
SHPO	Ohio State Preservation Officer
SIC	standard industrial classification
SME	Subject Matter Expert
SNM	special nuclear material
SPCC	Spill Protection Control and Countermeasures
SRD	System Requirements Document
SRP	Standard Review Plan
SSCs	structures, systems, and components
SWPP	Storm Water Pollution Prevention
TDAG	Training Development and Administrative Guide
TEDE	Total Effective Dose Equivalent
TLDs	Thermoluminescence Dosimeters
TLV	Threshold Limiting Value
TPQ	threshold planning quanitity
TQs	Threshold Quantities
TRM	Training Requirement Matrices
TWC	Tower Water Cooling
UL	Underwriters Laboratories
UPS	uninterruptible power supply
USA	Upper Suspension Assembly
USACE	United States Army Corps of Engineers
USEC	USEC Inc.
USGS	U.S. Geological Survey
USL	upper safety limit
UST	underground storage tank
VHRA	Very High Radiation Area
WCA	workers in the controlled area
WI/CL	What-if/Checklist
WRA	workers in the restricted area

DEFINITIONS

Heeling – The process for removing the residual quantity of uranium material that remains in a cylinder after routine evacuation procedures.

Natural Uranium – Any uranium-bearing material whose uranium isotopic distribution has not been altered from its natural occurring state. Natural uranium is nominally 99.283 percent ^{238}U, 0.711 percent ^{235}U, and 0.006 percent ^{234}U (by weight relative to total uranium element).

Normal Uranium – Any uranium-bearing material having a uranium isotopic weight distribution that can be described as being (1) 0.700 to 0.724 percent in combined ^{233}U plus ^{235}U; and (2) at least 99.200 percent in ^{238}U.

Frequencies
When audit, measurement, surveillance, and or other frequencies are specified in license documents (e.g., License Application, Fundamental Nuclear Material Control Plan, etc.), the definitions below apply, unless otherwise specified. For unspecified times, an extension of 0.25 times the period is used.

Interval Designation (Frequency)	Interval Between Consecutive Actions	Maximum Interval Between Consecutive Actions
Five-year	5 years to the day	5 years to the day (unless specifically stated otherwise)
Triennially	36 months	45 months
Biennially	24 months	30 months
Annually	365 days	456 days
Semiannually	184 days	230 days
Quarterly	92 days	115 days
Monthly	31 days	39 days
Weekly	7 days	9 days
Daily	24 hours	30 hours
Per Shift	12 hours	15 hours
Twice Each Shift	6 hours	8 hours

CHEMICALS AND UNITS OF MEASURE

CaF_2	calcium fluoride
cfs	cubic feet per second
Ci	curie
cm	centimeters
cm^2	square centimeter
dpm	disintegration per minute
DUF_6	depleted uranium hexafluoride
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)
MTU	metric tons uranium
pCi	picocurie (one-trillionth of a curie)
pCi/L	piocuries per liter
ppm	parts per million
psf	pounds per square foot
psi	pounds per square inch
REM	Roentgen Equivalent Man
SWU	separative work units
U_3O_8	depleted uranium oxide
UO_2F_2	uranyl fluoride
UF_6	uranium hexafluoride
V	volt

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
$\mu g/m^3$	micrograms per cubic meter
μ	micron or micrometer (one-millionth of a meter)
^{235}U	uranium-235
^{99}Tc	technetium

EXECUTIVE SUMMARY

This license application was previously submitted by Centrus Energy Corp. (Centrus), formerly known as USEC Inc. the applicant for a license to possess and use special nuclear, source and by-product material in the American Centrifuge Plant located in Piketon, Ohio, under the *Atomic Energy Act* of 1954, as amended, 10 *Code of Federal Regulations* (CFR) Parts 70, 40 and 30, and other applicable laws and regulations. A primary mission of the American Centrifuge technology is to provide the United States with a reliable and economical source of enriched uranium. Centrus is the parent company of the American Centrifuge Operations, LLC (ACO), which is the current assignee of a sublease for portions of the Portsmouth Gaseous Diffusion Plant (GDP) reservation from the U.S. Department of Energy (DOE) through the *Lease Agreement between the U.S. Department of Energy and United States Enrichment Corporation for the Gas Centrifuge Enrichment Plant* (GCEP lease Agreement). ACO(the Licensee) is a wholly owned indirect subsidiary of Centrus and is a limited liability company formed under the laws of Delaware.

Deployment of the American Centrifuge Plant supports the national energy security goal of maintaining a reliable and secure domestic source of enriched uranium. Through amendments to the *Atomic Energy Act*, Congress created and privatized the Corporation with the intention that USEC would, among other things, conduct research and development as required, evaluate alternative technologies for uranium enrichment and help maintain a reliable and economical domestic source of enriched uranium. Centrus continues that fundamental mission through its indirect subsidiary ACO (the Licensee).

The Licensee is responsible for the design, fabrication, installation, operation, maintenance, modification and testing of the American Centrifuge Plant. The American Centrifuge Plant is a uranium enrichment facility designed to enrich, safely contain and handle uranium hexafluoride up to 10-weight percent uranium-235. ACO currently holds a license for a term of 30 years. The initial modular design produces approximately 3.8 million separative work units annually. This submittal continues with modular deployment of the American Centrifuge Plant and the next phase of enrichment production, which involves deployment of a cascade of 16 centrifuges to demonstration production of high-assay, low-enrichment uranium fuel for advanced reactors. The design of the American Centrifuge Plant 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 American Centrifuge Plant is located on U.S. Department of Energy (DOE) owned land in rural Pike County, a sparsely populated area in south central Ohio. Some of these facilities are leased to the Licensee. The DOE reservation has been studied and characterized extensively by both DOE and Centrus, formerly USEC. The facilities to be utilized for the American Centrifuge Plant, which are part of the former DOE Gas Centrifuge Enrichment Plant program, were built in the early 1980s. The existing facilities will be refurbished to accommodate the American Centrifuge Plant. New facilities will be constructed to house withdrawal and product operations for the commercial American Centrifuge Plant. The commercial American Centrifuge Plant operation will also use other existing site-wide services such as laboratory analysis, fire protection, security, medical, 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 American Centrifuge Plant 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

This license application was previously submitted by Centrus Energy Corp. (Centrus), formerly known as USEC Inc., for the American Centrifuge Plant (ACP). It encompasses the construction, manufacturing, start-up, operations, maintenance, and decommissioning of a uranium enrichment facility using American Centrifuge technology that will produce approximately 3.8 million separative work units (SWU) annually.

The United States Enrichment Corporation leases portions of the Portsmouth Gaseous Diffusion Plant (GDP) reservation from the U.S. Department of Energy (DOE) through the *Lease Agreement between the U.S. Department of Energy and United States Enrichment Corporation for the Gas Centrifuge Enrichment Plant* (GCEP Lease Agreement). Pursuant to a 2006 amendment to that lease agreement, Centrus subleased space for the American Centrifuge Lead Cascade Facility (Lead Cascade) and the ACP from the United States Enrichment Corporation. Centrus, with approval of the DOE, assigned the sublease for the space for the ACP to the Licensee, American Centrifuge Operating, LLC (ACO). The Licensee and its agents will conduct activities within the leased facilities and access and egress thereto, in accordance with this license application.

The ACP utilizes existing buildings located on the DOE reservation near Piketon, Ohio, that were built to support the gaseous centrifuge process beginning in the 1980s, in addition to several newly constructed buildings and facilities.

The ACP is the third step in the plan to deploy the American Centrifuge technology. The first step was the centrifuge testing in Oak Ridge, Tennessee, to upgrade, and demonstrate an economically attractive gas centrifuge and enrichment process. The second step was the deployment of the Lead Cascade in Piketon, Ohio, which provided reliability, performance, cost, and other vital data on the ACP enrichment process. American Centrifuge technology is modular, with the basic building block of enrichment capacity being a cascade of centrifuges. Information gained and work performed during the centrifuge testing and Lead Cascade projects included vital information on performance, reliability, and economics that will be used in the construction of the ACP. A license application was prepared pursuant to the *Atomic Energy Act* of 1954 as amended, 10 *Code of Federal Regulations* (CFR) Parts 70, 40, 30, and other applicable laws and regulations. The commercial ACP operation is designed to enrich and safely contain and handle uranium hexafluoride (UF_6) up to 10-weight (wt.) percent uranium-235 (^{235}U). This license application includes the High Assay Low Enriched Uranium (HALEU) Demonstration Program which is designed to enrich and safely contain and handle UF_6 with an operational limit that is less than 20.0 wt. percent ^{235}U.

This license application follows the format and content guidelines provided in NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications*, Revision 2 (Reference 1). The information provided reflects the design in sufficient detail to enable a reviewer to make a definitive evaluation that the ACP can be constructed and operated without undue risk to the health and safety of the public and with no significant impact to the environment.

The ACP uses portions of the Portsmouth Gaseous Diffusion Plant (GDP) and the former DOE Gas Centrifuge Enrichment Plant (GCEP) along with eight new facilities. The ACP utilizes existing utilities and infrastructure that support the DOE reservation along with the utilities and infrastructure that were intended to support GCEP. New facilities are necessary for feed, withdrawal, sampling, and blending/transfer operations. Centrus has updated the American Centrifuge technology from that used in the GCEP program, but the American Centrifuge components remain compatible with existing infrastructure and buildings/facilities.

The HALEU Demonstration Program is a program awarded by DOE's Nuclear Energy Oak Ridge Site Office for the demonstration of the HALEU production to support DOE research and development (R&D) activities and programs. The HALEU Demonstration Contract was awarded on May 31, 2019 and definitized on October 31, 2019 (Reference 17). The two primary objectives of the HALEU Demonstration Program is for American Centrifuge Operating, LLC (ACO), the licensee, to deploy a 16-machine AC-100M HALEU cascade in the Piketon facility to produce 19.75% ^{235}U enriched product and to demonstrate the capability to produce HALEU utilizing US-origin uranium enrichment technology. The HALEU Demonstration will be deployed in a subset of the larger ACP with deviations noted as appropriate in the sections that follow.

It is the intent of the licensee to deploy portions of the ACP in a modular fashion to accommodate market demand on a scalable, economical gradation. This modular deployment will encompass utilization of cascades of LEU production for customer product or feed material into HALEU cascades.

1.1 Plant and Process Description

This section describes the buildings and facilities that comprise the ACP located on the DOE reservation in Piketon, Ohio, and describes the process by which the plant will operate. Facilities are those buildings and systems identified in the lease agreement between the United States Enrichment Corporation and DOE. The ACP buildings and facilities are grouped in two categories, primary and secondary in the Integrated Safety Analysis (ISA) Summary. Figure 1.1-1 (located in Appendix B) depicts the entire DOE reservation and the area where the ACP resides in the southwest quadrant. Figure 1.1-2 (located in Appendix B) depicts a closer view of the ACP area and shows the Primary and Secondary buildings. Primary facilities are those buildings or areas that could contain licensed material in quantities that could potentially result in consequences that exceed the performance criteria defined in 10 CFR 70.61 resulting from credible accidents or that directly control a primary facility. All other ACP facilities are considered to be secondary. A further description of primary and secondary facilities and a list of these buildings/facilities are in Sections 1.1.3 and 1.1.4 of this license application.

The uranium element appears in nature in numerous isotopes; the three major isotopes of interest have atomic weights of 234, 235, and 238. The ^{235}U isotopes are fissionable and capable of sustaining a critical reaction. Natural uranium contains 0.711 percent ^{235}U isotope. Isotopic separation processes separate uranium into two fractions, one enriched in the ^{235}U isotope, and the other depleted.

Prior to the enrichment process, uranium is combined with fluorine to form UF_6 from the uranium feed suppliers. The UF_6 arrives at the plant in a solid state and this UF_6 is sublimed from a solid to a gas and fed into the system. In the gas centrifuge process, the isotopic separation is accomplished by centrifugal force, which uses the difference in weight of the uranium isotopes to achieve this isotopic separation. UF_6 can be enriched up to 10 wt. percent assay ^{235}U in the commercial ACP operation. The plant withdraws the enriched (product) stream and the depleted (tails) stream in the gaseous state. The product and tails streams are then sublimed back into a solid state for handling and movement. The plant minimizes the amount of UF_6 in the liquid state.

Two process buildings are included in the initial deployment of the ACP to support a 3.8 million SWU production capacity with centrifuges arranged in cascades.

UF_6 feed to the HALEU Demonstration will be LEU UF_6 product with an enrichment of less than 5.0 wt.% ^{235}U. The HALEU Demonstration will enrich this material to an enrichment less than 20.0 wt.% ^{235}U in its product stream and will deplete the feed to a target tails stream enrichment of approximately equal to or less than 1.0 wt.% ^{235}U.

1.1.1 Site Boundary

The ACP is located approximately one and one half miles east of U.S. Route 23 on the approximately 3,700 acre DOE reservation. The area around the reservation is sparsely populated, with the nearest residential center located approximately four miles to the north of the reservation. The ACP is located in the southwest quadrant of the reservation and is situated on approximately 200 acres. The site boundary is the DOE reservation boundary, which is depicted in Figure 1.1-1 (located in Appendix B). Proximity of the ACP to the nearest member of the public (i.e., permanent residence) is about 2,200 feet (ft) [670 meters (m)].

1.1.2 Plant Layout

The ACP layout is depicted in Figure 1.1-1 in relationship to the DOE reservation and in Figure 1.1-2 (both located in Appendix B) for the ACP specifically. The ACP is comprised of various buildings/facilities and areas that house systems and equipment necessary to support the American Centrifuge uranium enrichment process. The ACP utilizes buildings and facilities that were part of GCEP, built in the early 1980s, part of the GDP that was built in the early 1950s, and newly constructed buildings and facilities. Descriptions of the major primary and secondary facilities are contained in the following sections. A brief listing of the buildings and facilities utilized for the ACP is located in Table 1.1-1.

The design of the plant complies with the performance requirements of 10 CFR 70.61, the Baseline Design Criteria specified in 10 CFR 70.64(a) and the defense-in-depth requirements contained in 10 CFR 70.64(b).

1.1.3 Primary Facilities Description

Primary facilities are those buildings/facilities or areas that could potentially contain licensed material in quantities that result in consequences that exceed the performance criteria defined in 10 CFR 70.61 resulting from credible accidents or directly controls a primary facility. The primary facilities directly involved in the enrichment process are the X-2232C Interconnecting Process Piping (IPP), X-3001 Process Building; X-3002 Process Building; X-3012 Process Support Building; X-3344 Customer Services Building; X-3346 Feed and Withdrawal Building; and X-3346A Feed and Product Shipping and Receiving Building. Other buildings and areas that provide direct support functions to the enrichment process are the X-7725 Recycle/Assembly Building; X-7726 Centrifuge Training and Test Facility; X-7727H Interplant Transfer Corridor;

X-745G-2 Cylinder Storage Yard; X-745H (future) Cylinder Storage Yard, and X-7746S, X-7746W Cylinder Storage Yards and Intraplant Roadways. These buildings and areas are where special nuclear material and hazardous material can be found and are considered to be the primary facilities in their functional support of the uranium enrichment process. A description of the primary facilities and their function is provided in the following sub-sections and are listed and briefly described in Table 1.1-1. An overall depiction of the enrichment processes is provided in Figure 1.1.3-1 located in Appendix E.

ACO's long-term goal is to resume commercial enrichment production consistent with market demand. The ACP design is modular, with the basic building block of enrichment capacity being a cascade of centrifuges. Modular deployment would accommodate market demand on a scalable, economical gradation. The Fire Safety Program will be implemented to support the modular deployment, such that the fire protection systems/services are in place when needed.

The next phase of enrichment production includes the deployment of a cascade of 16 centrifuges to demonstrate production of high-assay, low-enriched uranium (HALEU) fuel for advanced reactors. The primary building/facilities directly involved in HALEU Demonstration are the X-3001 Process Building, X-3012 Process Support Building, X-7725 Recycle/Assembly Building, X-7726 Centrifuge Training and Test Facility, and X-7727H Interplant Transfer Corridor. It is also noted that HALEU Demonstration does not involve or include the use of any liquid UF_6 handling operation or those facilities.

1.1.3.1 X-3001 and X-3002 Process Buildings

The initial deployment of the ACP includes two process buildings, which are located in the southwest quadrant of the DOE reservation: X-3001 and X-3002. The primary purpose of the process buildings is to house the centrifuges and support systems necessary to perform the actual enrichment process. Both buildings are similar in construction, layout, and design. Each building is approximately 416 feet (ft) by 730 ft (approximately 304,000 square feet [ft^2]) and has a large high bay process area and two utility areas. The height of each building is approximately 87 ft in the high bay area and 49 ft in the utility areas. The nearest reservation boundary is 2,606 ft to the west of the X-3001 building. Figure 1.1-3 (located in Appendix B) depicts the typical equipment and process flow for the X-3001 and X-3002 buildings. Figures 1.1.3.1-1, 1.1.3.1-2, 1.1.3.1-3, and 1.1.3.1-4 (located in Appendix E) also depict the equipment layout for the X-3001 and X-3002 buildings.

At the north and south ends of X-3001 and X-3002 buildings are equipment/utility bays and mezzanines where auxiliary equipment is housed. Items in these areas consist of heating and ventilation equipment, cooling water pumps, vacuum pumps, electrical switchgear, and standby electrical equipment (i.e., diesel generators, battery rooms, and uninterruptible power supply [UPS] systems). Building vents for the purge and evacuation vacuum systems are also located in the buildings. The vents are monitored and are permitted through the Ohio Environmental Protection Agency (OEPA).

The east side of the X-3001 building is connected to the X-3012 building, which is connected to the west side of the X-3002 building. The X-7727H corridor is connected to the west side of the X-3001 building. The X-2232C piping connects to the southwest corner of the X-3001 building at a valve house where it both enters and exits the building. The connection of the X-2232C piping exits the east side of the X-3001 building and enters and exits the X-3002 building on the west side through a valve house as well.

The centrifuges are installed in the high bay area in a cascade arrangement. The cascades are supplied UF_6 feed from a header from the Feed Area in the X-3346 building. The centrifuges in each cascade are grouped into stages that are connected in series. The feed, product, and tails lines to and from each centrifuge within a stage connect into stage headers that convey the UF_6 streams between stages. The depleted material from the bottom stage is piped through the X-2232C piping to the X-3346 building Withdrawal Area to be withdrawn as tails. The enriched material from the top stage is piped through the X-2232C piping to the X-3346 building Withdrawal Area to be withdrawn as product. For commercial ACP operations the cascade enrichment is normally less than 5.5 wt. percent ^{235}U, but enrichment levels up to 10 wt. percent ^{235}U are allowable.

The HALEU Demonstration cascade utilizes a similar centrifuge design to that used for the Lead Cascade. The equipment necessary to perform the enrichment process is in the X-3001 Process Building and consists of product and tails withdrawal system, uranium hexafluoride (UF_6) cylinders, centrifuges, and supporting units. The product and tails withdrawal systems use three cold boxes. NaF traps are used for additional withdrawal capacity during dumping. A 30B UF_6 cylinder is used for the feed material. Centrifuges and supporting units are placed in the Train 3 area of the X-3001. For further plant and process specifics related to the HALEU Demonstration Program, refer to LA-3605-0003A, *Addendum 1 of the Integrated Safety Analysis for the American Centrifuge Plant – HALEU Demonstration* (Reference 7).

1.1.3.2 X-3012 Process Support Building

The X-3012 houses the operational area, maintenance area, and the transfer aisleway that services the X-3002 building. The X-3012 building is located between the X-3001 and X-3002 buildings. The X-3012 building, which is approximately 201 ft by 240 ft at grade level, has a ground floor area of approximately 48,000 ft^2, and has a total covered floor space area of approximately 56,200 ft^2, which includes the ground floor and two mezzanine areas. The transfer aisle way between the X-3001 and X-3002 and through the X-3012 building measures 30 ft wide by approximately 59 ft high by 200 ft long and divides the building into north and south sections. The north section is approximately 17 ft high and contains the operational area. The south section of the building is approximately 26.5 ft high and contains the maintenance areas. The nearest reservation boundary is 3,024 ft to the west of the X-3012 building.

The X-3012 building is divided into three functional areas: an operational area, maintenance area, and a centrifuge transfer aisleway. The operational area is located in the north section of the building and includes the Area Control Room (ACR) for the X-3001 and X-3002 buildings; offices; lunchroom; restrooms; battery room; switchgear room; and heating, ventilation, and air conditioning (HVAC) rooms. A mezzanine above the north section contains the mechanical equipment room for the building. The ACR provides the central operating functions to monitor and control both the X-3001 and X-3002 building centrifuges and processes. The maintenance area is located in the south section of the building and includes: maintenance shops, storage areas, a battery charging room, offices, men's and women's locker rooms, restrooms, and a mezzanine area with additional office areas and HVAC rooms. The X-7727H corridor is used for the transport of centrifuges into and out of the X-3002 building.

Access between the X-3001 and X-3002 buildings is provided via the transfer aisleway, which also provides access between the operational and maintenance areas of the X-3012 building.

1.1.3.3 Feed, Withdrawal, and Product Operations

Figure 1.1-4 (located in Appendix B) depicts a process flow schematic of Feed, Withdrawal, and Product operations.

1.1.3.3.1 X-3346 Feed and Withdrawal Building

The X-3346 building is located in the southwest quadrant of the DOE reservation. The X-3346 building is located approximately 1,000 ft south-southwest of the X-3001 building. The nearest reservation boundary is 1,865 ft to the west of the X-3346 building. The X-3346 building is connected to the X-3001 and X-3002 buildings by the X-2232C piping to provide UF_6 feed to the enrichment process and for the withdrawal of product (enriched) and depleted (tails) UF_6 material.

The X-3346 building has a covered floor area of approximately 154,000 ft^2 with two distinct areas of operation to meet process feed, blending/transfer requirements and product and tails withdrawal. The X-3346 building has two distinct areas of operation. The first area, referred to as the Feed Area, supports the front end of the overall enrichment process by housing the equipment necessary to provide UF_6 feed. This area also houses the equipment necessary to blend/transfer UF_6 between cylinders, including filling customer cylinders. The second area, referred to as the Withdrawal Area, supports the back end of the enrichment process by housing the equipment necessary to withdraw enriched UF_6 into cylinders and to withdraw depleted UF_6 (tails) into tails cylinders. Figures 1.1-5a, 1.1-5b, 1.1-5c, 1.1-5d and 1.1-5e (located in Appendix B) depict the typical equipment and process flow for the X-3346 building. Figures 1.1.3.3.1-1, 1.1.3.3.1-2, and 1.1.3.3.1-3 (located in Appendix E) also depict the equipment layout for the X-3346 building.

The Feed Area of the X-3346 building houses electrically heated feed ovens. UF_6 feed is processed through freezer/sublimers to purify the feed material before being fed into the process manifolds/piping. There are separate manifolds that direct each stream to the X-3001 and X-3002 buildings through the X-2232C piping. The light gases removed during the feed purification process are evacuated to an evacuation system in the X-3346 Withdrawal Area. The Feed Area also houses the dedicated feed ovens and cold boxes required to perform blending/transfer operation between the cylinders. See Figure 1.1.3.3.1-4 (located in Appendix E) for a typical depiction of a cold box. This includes filling customer cylinders. A capability is provided to transfer UF_6 from the feed ovens to Withdrawal Area for blending of enriched UF_6 from the enrichment process. The Feed Area has accountability scales for weighing the feed and other cylinders. The location of the feed ovens and cold boxes provides the cylinder transporter sufficient room to transport the UF_6 cylinders between rows of ovens. The cylinder transporters move the cylinders into and out of the feed ovens and cold boxes.

The X-3346 building Withdrawal Area houses the equipment that functions to withdrawal enriched and depleted UF_6 from the process. Product (enriched UF_6) withdrawal is performed via the use of trains of vacuum pumps which directly transfer UF_6 at sub-atmospheric pressures and desublime the UF_6 into cylinders located in cold boxes. These cylinders may be customer cylinders. Different product assays can be withdrawn to the X-3346 building Withdrawal Area from the X-3001 and X-3002 buildings and blending of the material withdrawn may be blended with feed material. Tails withdrawal is performed via the use of multi-stage compressor trains which perform the withdrawal at sub-atmospheric pressures and then desublime the depleted UF_6 into tails cylinders located in cold boxes. A surge drum is in-line ahead of the tails compressor trains and a surge drum is in-line behind each of the two tails compressor trains. The Withdrawal Area has accountability scales for weighing the cylinders. The location of the cold boxes provides the cylinder transporter sufficient room to transport the UF_6 cylinders between rows of cold boxes. The cylinder transporters move the cylinders into and out of the cold boxes.

The primary specialized support systems for the Feed and Withdrawal Area are those associated with purge and evacuation; these systems are located in the X-3346 Withdrawal Area and support operations in the X-3344 building as well. These support systems service both process lines and equipment and local area UF_6 "wisp" management systems that control small UF_6 releases that might occur during operations (i.e., disconnecting pigtails from cylinders). Banks of cold traps are used to remove UF_6 from the gas streams before the gas is transferred though chemical traps and then to a vent through blowers. The purge and evacuation vents are monitored and permitted through the OEPA. Other major support equipment includes refrigeration units, precision scales, and bridge cranes. Other auxiliaries are those that are customary (e.g., electrical supply, instrument air, cooling water, etc.).

1.1.3.3.2 X-3346A Feed and Product Shipping and Receiving Building

The X-3346A building is located in the southwest quadrant of the DOE reservation approximately 300 ft south of the X-3346 building. The building measures approximately 100 ft in width, 40 ft in height, and 190 ft in length with a covered floor area of approximately 19,000 ft^2. This building serves as the focal point for the receipt and shipping of natural and enriched uranium in U.S. Department of Transportation (DOT) approved cylinders and Protective Shipping Packages (PSPs), as required. The nearest reservation boundary is 1,820 ft to the west of the X-3346A building. Figure 1.1-6 (located in Appendix B) depicts the typical equipment and process flow for the X-3346A building. Figure 1.1.3.3.2-1 (located in Appendix E) also depicts the equipment layout for the X-3346A building.

The X-3346A building is connected to the X-3346 building by a bridge crane rail system that serves both the X-3346 and X-3346A buildings. X-3346A has doors on the north and south sides of the building for either trucks (tractor trailer) or cylinder handling equipment or cranes utilized for movement of cylinders.

The X-3346A building contains the operations associated with receiving full UF_6 feed cylinders and returning empty feed cylinders to vendors and the receipt of empty product cylinders and shipment of full product cylinders to customers. The building includes a large shipping and receiving area, cylinder staging area, offices, and a trucker's rest area.

1.1.3.3.3 X-3344 Customer Services Building

The X-3344 building is located in the southwest quadrant of the DOE reservation to the southwest of the X-3001 building and to the north of the X-3346 building. The building is single story and has a covered floor area of approximately 35,200 ft^2 with one area of operation to meet the process sampling requirements. The nearest reservation boundary is 2,780 ft to the west of the X-3344 building. Figure 1.1-7 (located in Appendix B) depicts the typical equipment and process flow for the X-3344 building. Figure 1.1.3.3.3-1 (located in Appendix E) depicts the equipment layout for the X-3344 building. See Figure 1.1.3.3.3-2 (located in Appendix E) for a typical depiction of an autoclave.

The X-3344 Customer Services Building is the only building where liquid UF_6 may be present and a containment barrier (autoclave) is provided should an accident occur during sampling activities. The cylinders are enclosed in containment autoclaves when the UF_6 is in the liquid phase, to minimize the potential for a release of liquid UF_6. In the Customer Services Building, the basic approach to operations is to liquefy the UF_6 contained in cylinders within a closed autoclave, sample the liquid using a sample manifold and sample cylinders within the autoclave, then allow the cylinders to cool until the UF_6 has re-solidified. Cooling capability is supplied to expedite the cool-down process and shorten the cycle time on each individual autoclave. Any approved UF_6 container (2.5-ton, 10-ton or 14-ton) may be heated in an electrically heated containment autoclave for sampling purposes. There are no UF_6 process lines that are external to the autoclaves; the piping used for evacuation is disconnected from the cylinder and sample manifold prior to closure of the autoclave and contains only trace quantities of UF_6.

The primary specialized support systems are those associated with evacuation. These support systems service both evacuation piping lines and equipment and local area UF_6 "wisp" management systems that control small releases that might occur during operations (i.e., disconnecting pigtails from cylinders). The evacuation piping is connected to the evacuation system in the X-3346 Withdrawal Area. The vent(s) are monitored and permitted through the OEPA. Other major support equipment includes feed ovens (heating and refrigeration units), precision scales, and bridge cranes. Other auxiliaries are those that are customary (e.g., electrical supply, instrument air, cooling water, etc.).

1.1.3.4 X-7725 Recycle/Assembly Building

The X-7725 building is located in the southwest quadrant of the DOE reservation. The X-7725 building is connected to X-7726 facility and the X-7727H corridor and is located to the north of the X-3001 and X-3002 buildings. The X-7725 building is approximately 540 ft x 820 ft (approximately 442,800 ft^2 area), and it contains a total floor space of about 837,900 ft^2 on five floors. The nearest reservation boundary is 2,431 ft to the west of the X-7725 building. Figure 1.1-8 (located in Appendix B) depicts the typical equipment and process flow for the X-7725 building and its relationship to X-7726 facility and the X-7727H transfer corridor. Figures 1.1.3.4-1 and 1.1.3.4-2 (located in Appendix E) also depict the equipment layout for the X-7725 building.

The purpose of the X-7725 building is to provide an area where centrifuges can be manufactured, assembled, tested, and maintained. The assembly of centrifuges begins with receipt of centrifuge components. Then these components are stored and staged for assembly. Centrifuge components and subassemblies are assembled into a complete centrifuge on one of the centrifuge assembly stands.

If some of the centrifuges are assembled faster than can be transported for installation, these centrifuges can be stored in the buffer storage area. Some completely assembled centrifuges are tested in the Gas Test stands using UF_6 to verify the correct placement of centrifuge components and the proper operation of the centrifuge. The Gas Test is performed in the X-7725 building prior to moving the centrifuges to the process building for installation. Drawing X-7725-0003-ME (located in Appendix A) depicts the Gas Test process flow.

There are various support areas throughout the building on each level. These areas include cranes; mechanical equipment rooms; electrical equipment rooms; freight and personnel elevators; HVAC equipment rooms; maintenance areas; offices; restrooms; shower/locker rooms; shipping/receiving/materials storage areas; and other material handling equipment.

An overhead crane system traverses the buffer storage area and assembly area of the X-7725 building for movement of centrifuges or other large components.

1.1.3.5 X-7726 Centrifuge Training and Test Facility

The X-7726 facility is located in the southwest quadrant of the DOE reservation. The X-7726 facility is connected and adjacent to the northwest corner of the X-7725 building. The X-7726 facility has an overall height of approximately 80 ft, contains approximately 28,000 ft^2 of floor space at ground level and contains a total of 49,500 ft^2. The nearest reservation boundary is 2,431 ft to the west of the X-7726 facility. Figure 1.1-8 (located in Appendix B) depicts the typical equipment and process flow for the X-7726 facility and its relationship to X-7725 building and the X-7727H corridor.

The facility was originally built to support training of plant personnel for centrifuge assembly and testing. This facility will initially be used for centrifuge component manufacturing and centrifuge assembly, and then primarily used for a centrifuge assembly training and centrifuge component preparation area for the ACP.

The X-7726 facility is an area where material and components are received; components or subassemblies are inspected and tested; the components are assembled as centrifuges; the final assembly is evacuated and leak checked; and repairs are performed to the centrifuge or subassemblies until the X-7725 building is available for use. Then these functions will be performed in the X-7725 building. The X-7726 facility will then be used as a backup manufacturing/assembly area and may also be used for select repair of failed centrifuges or for disassembly of failed centrifuges for failure analysis. The X-7726 facility will continue to be used as a training area for centrifuge subassembly preparation, column assembly, and centrifuge assembly.

An overhead crane system traverses the length of the X-7726 facility for movement of centrifuges or other large components.

There are various support areas throughout the building to provide the necessary ancillary support for the centrifuge assembly operations and personnel. These areas include mechanical equipment rooms; electrical equipment rooms; freight and personnel elevators; HVAC equipment rooms; maintenance areas; offices; restrooms; and shower/locker rooms.

1.1.3.6 X-7727H Interplant Transfer Corridor

The X-7727H corridor is located in the southwest quadrant of the DOE reservation. The nearest reservation boundary is 2,480 ft to the west of X-7727H corridor. The X-7727H corridor measures approximately 30 ft in width, 59 ft in height, and 750 ft in length. There are 55 ft by 25 ft doors located where the corridor meets the X-7725 building and X-3001 building. Figure 1.1-9 (located in Appendix B) depicts the typical equipment and process flow for the X-7727H building.

The X-7727H corridor is an elongated structure that connects the X-7725 building with the X-3001 building. It provides a protected pathway to transport centrifuges from the X-7725 building or X-7726 facility to the process buildings or back as necessary. The X-7727H corridor also serves as a shipping and receiving area for equipment and components during construction and operation activities. At the south end of the corridor is a smaller structure/service area, known as the service module unloading area.

1.1.3.7 Cylinder Storage Yards (X-745G-2, X-745H, X-7746S, and X-7746W)

The uranium enrichment process relies on the use of cylinders to allow movement and storage of UF_6 material outside of the process. This method of material handling requires storage areas for cylinders. The ACP cylinder yards provide this storage for natural feed uranium, depleted (tails) uranium, and enriched (product) uranium awaiting shipment. UF_6 cylinders may be stored in any storage yard regardless of use, although cylinders of a certain type may be routinely stored in a particular yard. Figure 1.1-2 (located in Appendix B) depicts the ACP layout and depicts the location of the various cylinder yards.

There are four cylinder storage yards that support the ACP. Two of the yards are located adjacent to the X-3346 building (X-7746S and X-7746W yards), and the other two yards are located just north of the reservation Perimeter Road to the north of the GDP X-344 UF_6 Sampling Facility (X-745G-2 and X-745H yards). The X-7746S, X-7746W, and X-745G-2 Cylinder Storage Yards provide approximately 47,000 ft^2, 132,000 ft^2, and 135,000 ft^2, respectively. The nearest reservation boundary is to the west approximately 1,982 ft from the X-7746S and W Cylinder Storage Yards, and 2,827 ft from the X-745G-2 Cylinder Storage Yard. The Cylinder Storage Yards are designed primarily for storage of 2.5-ton, 10-ton, and 14-ton UF_6 cylinders.

1.1.3.8 X-2232C Interconnecting Process Piping

The X-2232C piping is any process piping that is external to the primary facilities. The X-2232C piping is the piping that connects the X-3346 building to the X-3001 building and the X-3002 building to the X-3001 building to provide feed to the X-3001 and X-3002 buildings and return product and tails to the X-3346 building. The nearest reservation boundary is 2,225 ft to the west of the X-2232C piping. Figure 1.1-10 (located in Appendix B) depicts the typical equipment and process flow for the X-2232C piping.

The X-2232C piping is typically located in a series of elevated enclosures or modules that run from the X-3346 building Feed Area to the X-3001 building valve house (approximately 1,700 ft) and then to the X-3002 valve house (approximately 800 additional ft) to provide feed for enrichment. The X-2232 C piping also runs in the reverse direction from the X-3002 valve house then from the X-3001 valve house to the X-3346 Withdrawal Area for withdrawal of enriched and depleted UF_6. The standard X-2232C piping module is approximately 40 ft long. Some piping modules are of non-standard lengths or shapes to accommodate vertical loops to give extra clearance across roadways and to fit-up to buildings. The X-2232C piping enclosures are insulated to minimize heat loss and heated to prevent the freeze-out of UF_6.

1.1.3.9 X-2202 Roads

No highways enter the DOE reservation. There are access roads that intersect with the Perimeter Road from four directions.

The reservation where the ACP is located has an extensive roadway system. The buildings/facilities on the reservation are serviced with a system of roads, which as a rule generally follow a north-south grid. The volume of traffic on the reservation is low and traffic is limited. Most plant personnel are required to use parking adjacent to the portals. The roadways allow for easy and safe movement of people, equipment, and material.

1.1.4 Secondary Facilities Description

In addition to the primary facilities, there are a number of secondary buildings/facilities and areas that provide indirect support to the ACP enrichment process. No special nuclear material, natural uranium, depleted uranium, or other hazardous radiological materials are found in these buildings/facilities and areas. The support buildings include various electrical utilities, fire protection, sewage treatment, water treatment, hot water production, compressed air, and others. However, some of the utilities and support services are procured. Utilities procured by the ACP include high voltage electrical power, firewater, sanitary water, sanitary sewer, communications, and non-potable cooling water. Support services procured by the ACP include emergency response and administrative support. The procured utilities and services are provided through existing buildings and services.

The major secondary buildings/facilities are depicted in Figures 1.1-1 and 1.1-2 (both located in Appendix B) and include the X-6000 Cooling Tower Pump House, Air Plant, and Air Plant Support Systems; X-6002 Boiler System; X-6002A Oil Storage Facility, X-7721 Maintenance, Stores and Training Building, X-7725A Waste Accountability Facility, and X-7745R Recycle/Assembly Storage area, respectively. A brief description of the major secondary facilities and their functions along with some major public warning and security systems are provided in the following sub-sections.

1.1.4.1 Section Reserved For Future Use

1.1.4.2 X-220E1 and X-220E3 Evacuation Public Address System

The Evacuation Public Address (PA) System is in place to provide instructions or notification in the event of an incident requiring evacuation or sheltering of reservation/plant personnel. The X-300 Plant Control Facility (PCF) PA system control console is continuously manned. During emergencies, the PA system is not used for routine traffic. The PA system serves most occupied plant buildings/facilities.

1.1.4.3 X-220R Public Warning Siren System

The Public Warning Siren System is used to provide notification to the public within a two-mile radius of the DOE reservation in the event of an incident requiring evacuation or sheltering of the public. The system is comprised of sirens on poles/towers around the two-mile radius and an electronic siren controller at the X-300 PCF, X-1020 Emergency Operations Center, and local sheriff's department.

1.1.4.4 Electrical Distribution Systems

Electrical power is supplied from the external 345 kilovolts (kV) power grid through the X-530A Switchyard to the X-5001 Substation via the X-5015, 345 kV Underground Cable. The X-5001B and X-530G oil pumping stations are the facilities that make up the high pressure oil system that provides the necessary dielectric medium for the underground cable. At the X-5001 Substation, the electrical power is stepped down in voltage to 13.8 kV, via the 345 kV to 13.8 kV power transformers. The power transformers are protected by the X-5001A Valve House that supplies water to the power transformer deluge system. Electrical power enters the X-5000 Switch House via the bus duct from the power transformers. Power is distributed throughout the ACP by the X-2215A Underground Electrical Distribution to Process Buildings and X-2215B Electrical Distribution to Areas Other Than Process Buildings. The distribution voltages are further stepped-down as necessary, depending on the building or facility requirements to power items (i.e., centrifuges, pumps, compressors, cranes, elevators, lighting, HVAC, and offices). The X-2215C Exterior Lighting Fixtures provides exterior lighting for streets and fences throughout the ACP.

Most buildings and facilities are provided with double-ended service, wherein two substations supply power to switchgear separated by a tiebreaker. If one transformer fails or requires servicing, the entire building or facility load can be transferred to the remaining unit. Normally the transformers comprising a double-ended unit are fed from different switchyard busses.

Certain 480 V and 208 V substations are equipped with standby power in the form of diesel engine generators. The purpose of the diesel generators are to maintain power to essential systems in the event normal power is lost or interrupted to these systems momentarily or for long periods of time.

Standby power is provided by diesel engine driven generators in situations where a loss of normal power cannot be interrupted without causing damage to equipment or hazards to personnel. Single backup power is supplied by a standby generator to those systems for which power outages would result in potential damage to equipment, or substantial delays in restoring normal operations after an extended outage. Following a loss of normal power, standby generators will automatically start and pickup essential loads within a prescribed amount of time.

1.1.4.5 Section Reserved For Future Use

1.1.4.6 X-2220N Security Access Control and Alarm System

Due to the classified and proprietary nature of the ACP activities and equipment, access to areas classified as Security Areas and Vault-type Room(s) is controlled utilizing a Security Access Control and Alarm System. The system consists of two distinct subsystems: an Intrusion Detection System (IDS) and an Access Control System (ACS). The IDS provides interior protection and the ACS provides high-security entry controls. The two subsystems report to a single operator's workstation forming a single security system.

1.1.4.7 Security Fencing and Portals

The ACP is within a secured fenced area. This area consists of approximately three and a half miles of eight ft high chain-linked fence and barbed wire encompassing approximately 200 acres of the southwest quadrant of the Controlled Access Area (CAA). Various gates support normal operation and provide emergency egress. The fence is routinely patrolled and is well maintained.

Access to the ACP CAA consists of portals and gates at specific locations. When in use, portals are either staffed and gates (when open) are under surveillance by Protective Personnel with communications equipment or the portals are equipped with rotogates with an electronic badge reader. Portals are secured with high security locks when not in use. Signs are posted at the CAA access portals and gates identifying contraband items that are not permitted within the CAA without specific approval. Illumination is in place at the CAA access portals and gates to assist Protective Personnel and building or plant personnel in detecting unauthorized persons and to permit examination of badges and vehicles. In the event of extended power outages where necessary illumination is compromised, compensatory measures (e.g., standby lighting) are implemented.

CAA portal and gate operations are further defined and locations identified in the *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant.*

1.1.4.8 X-6000 Cooling Tower Pump House, Air Plant, and Air Plant Support Systems, and X-6001 Cooling Tower

The X-6000 Cooling Tower Pump House, Air Plant, and Air Plant Support Systems is located east of the X-3002 building and is approximately 223 ft long and 80 ft wide. The building contains two distinct sections: Cooling Tower Pump House and the Air Plant. The Air Plant is located at the north end section and the Cooling Tower pump equipment is located at the south end section of the X-6000 building. The X-6000 building contains the necessary equipment/systems to distribute dry compressed air to the ACP and to provide the requisite water to the X-6001 Cooling Towers for the removal of heat from the process buildings.

The X-6001 tower is located west of the X-1007 Fire Station and is approximately 100 ft east of the X-6000 building. The X-6001 tower measures approximately 282 ft long, 55 ft wide at the base, and is approximately 24 ft high from grade to upper deck, consisting of five cells. The X-6001 tower also contains the necessary equipment/systems, fans, piping, and hardware structures to satisfy the necessary cooling requirements for the process buildings.

1.1.4.9 X-6002 Boiler System

The X-6002 system is a gas-fired boiler system located between the X-6002A Oil Storage Facility and the X-7721 building just northeast of the X-3002 building. The boiler system provides hot water for heating.

The X-6002A facility is located east of the X-3002 building. The X-6002A facility supplies fuel oil to the X-6002 system when required. The boiler normally is operated on natural gas, but can use fuel oil as an alternate fuel.

1.1.4.10 X-7721 Maintenance, Stores, and Training Building

The X-7721 building is a multiple level building with approximately 138,000 ft^2 of total floor area. The purpose of the X-7721 building is to provide areas for maintenance shops; stores and receiving activities; and training.

1.1.4.11 X-7725A Waste Accountability Facility

The X-7725A facility is located in the southwest quadrant of the DOE reservation north of the X-7725 building and has approximately 29,400 ft^2 of floor space. This facility serves as a storage area for equipment and parts necessary for the maintenance and repair of the process and process support equipment.

1.1.4.12 X-7745R Recycle/Assembly Storage

The X-7745R storage area is a concrete pad immediately adjacent to and east of the X-7725 building providing approximately 215,200 ft^2 of space. This area is used mainly for clean, non-contaminated, outside, horizontal rack storage of centrifuge casings prior to being moved inside the building for centrifuge assembly. Other centrifuge components and miscellaneous storage may also be temporarily stored in this area.

1.1.4.13 X-2230B Sanitary Sewer

The X-2230B Sanitary Sewer system is an underground sewage collection system that through a series of piping and lift stations collects raw sewage from the ACP site and routes it to the DOE owned X-6619 Sewage Treatment facility. This facility is a NPDES permitted facility.

1.1.4.14 X-2230C Storm Sewer

The X-2230C Storm Sewer system is an underground drainage system to collect surface water from the ACP site. The water is routed through a series of piping to two holding ponds identified as X-2230N and X-2230M, both of which are NPDES permitted outfalls. This water is monitored for contaminants before being discharged into the nearby creeks.

1.1.5 Process Description

This process description is organized into eight sections that describe the gas centrifuge processes: 1) centrifuge program history; 2) separation fundamentals; 3) centrifuge fundamentals; 4) enrichment process theory; 5) total process configuration; 6) enrichment process support systems; 7) centrifuge assembly and movement systems; and 8) plant support systems. Additional details are provided in the ISA Summary.

1.1.5.1 Centrifuge Program History

For commercial production of uranium enriched in the ^{235}U isotope, a limited number of separation processes appear to be viable with technology currently available. In the United States, the electromagnetic process, gaseous diffusion process, and gas centrifuge process have been the primary methods employed since the inception of the uranium enrichment program during the Manhattan Project.

The gas centrifuge uranium enrichment program in the United States began in 1941. During World War II, the calutron and the gaseous diffusion processes were developed into viable techniques for producing enriched uranium more rapidly than the centrifuge process. As a result, work on the gas centrifuge technology was stopped. Development of centrifuge technology continued outside of the United States Government program until the Atomic Energy Commission resumed research and development work in 1960 at the Oak Ridge GDP under management of Union Carbide Corporation. Development progressed to the point that President Carter announced the switch from a GDP addition already under construction in Piketon, Ohio, to the more energy-efficient centrifuge process. The X-3001, X-3002, X-7726, and X-7725 buildings/facilities had been constructed by the time the GCEP program was cancelled in 1985. Six complete cascades were operating in parallel at the time of cancellation.

In 1993, the United States Enrichment Corporation took over uranium enrichment operations from the DOE at the GDP. It was recognized at that time that a newer more efficient separation technology ultimately would have to be deployed to replace the aging GDPs. After research on various separation technologies, USEC decided to deploy the American Centrifuge technology in 2002.

1.1.5.2 Separation Fundamentals

The processing of UF_6 into an isotopic content that enables 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 up to 10 wt. percent assay in the commercial ACP operation. The enrichment process in the HALEU Demonstration will increase the enrichment from a feed enrichment of up to 5.0 wt.% ^{235}U up to a target enrichment of 19.75 wt.%. The balance of uranium consists primarily of the ^{238}U isotope.

There is one methodology of enrichment commercially employed, the gas centrifuge process. This process consists of the interconnection of multiple "separation elements" in configurations known as cascades. Figure 1.1-11 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. Separative work is a descriptive mathematical quantity that measures the amount of effort required to effect the separation and is measured in Separative Work Units (SWUs).

1.1.5.3 Centrifuge Fundamentals

Figure 1.1-12 shows a simplified schematic of a gas centrifuge. A centrifuge 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 a stationary cylinder, called the casing, which maintains the rotor in a vacuum and provides physical containment of components in the unlikely event of a major centrifuge failure. Other major components of a centrifuge include upper and lower suspension systems, and a column.

Figure 1.1-12 depicts a modern centrifuge. The outer casing is at a high vacuum to minimize the drag on the high-speed rotor. Feed enters the centrifuge approximately mid-way down the column and mixes with the up flowing process gas layer near the rotor wall. The lighter component (enriched) stream flows upward where a scoop, positioned near the rotor wall, withdraws the enriched stream. The remaining portion of the gas stream flows down the wall, becoming the depleted stream where a scoop, positioned near the rotor wall, similarly withdraws the depleted stream.

The separation capacity of a centrifuge is a function of the difference in the assay at the top and bottom of the rotor. 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 separation unit (centrifuge) is higher than that of the gaseous diffusion separation element (converter). Due to the higher separation factor of the centrifuge separation unit, there are fewer stages required in a centrifuge cascade than in a gaseous diffusion cascade. However, the production rate for a single centrifuge separation unit is much less than a gaseous diffusion separation unit. Therefore, it is necessary to operate multiple centrifuge separation units in parallel in order to achieve production levels.

The high vacuum and partially armored casing serves two key functions: to minimize drag and confine the potential debris generated from a rotor failure while operating. The current centrifuge design relies on a molecular pump on each centrifuge backed-up by a mechanical vacuum pump to maintain this high vacuum in the casing. The primary function of the vacuum system is to remove any traces of gases that escape from the rotor through the column gap or atmospheric leaks from the casing seals.

Centrifuges are arranged in parallel to make-up a stage. The centrifuges in a stage receive a common feed and discharge enriched material and depleted material into common headers. Stages are then arranged in series to make-up a cascade. The inter-stage flow arrangement is depicted schematically in Figure 1.1-13 for a typical cascade. Each stage is represented by a single centrifuge, but the concept is that the enriched stream of the lower stage is set to closely match the assay of the external cascade feed and the depleted stream of the upper stage is also set to closely match that assay. The lower stage depleted stream header is the cascade tails header and the upper stage enriched stream header is the cascade product header.

1.1.5.4 Enrichment Process Theory

To produce enriched uranium at the desired ^{235}U assay, separation units are connected in series to form an enrichment cascade. Multiple cascades may be connected in parallel in order to produce enough product material of a given assay to meet customer orders.

1.1.5.5 Total Process Configuration

Total process configuration refers to how the enrichment process is carried out from the time natural uranium is received until finished product and process waste is shipped off-site. The process is divided into eight operations: 1) receipt of UF_6; 2) feeding of UF_6 into the enrichment process; 3) actual enrichment process, where the UF_6 assay is increased to its desired enrichment; 4) material withdrawal, where enriched and depleted UF_6 is removed from the enrichment process; a capability to withdraw feed material into product withdrawal to blend is also provided; 5) UF_6 sampling, where enriched UF_6 is sampled to ensure it meets customer specifications are met in either customer or source cylinders; feed, tails and dump cylinders are also sampled as required; 6) blending/transfer of enriched UF_6 between cylinders to fulfill customer specifications by sublimation and desublimation; 7) loading of UF_6 cylinders for shipment to customers; and 8) waste handling from waste generated from the entire process. See Figure 1.1-4 (located in Appendix B) and Figure 1.1.3-1 (located in Appendix E) for a functional depiction of the overall enrichment process.

1.1.5.5.1 Receiving Operations

The X-3346A building is the usual receiving point for cylinders. UF_6 feed cylinders, cylinders containing enriched product (such as Russian LEU material), customer shipping cylinders and overpacks, as well as, new and cleaned empty cylinders are received on-site via the X-3346A. Full feed cylinders (10- and 14-ton), customer cylinders (2.5-ton), and overpacks with customer cylinders are off-loaded, weighed, paperwork checked, and then the cylinders and overpacks are transferred to the appropriate storage areas until needed (see Figure 1.1-4 [located in Appendix B] for functional depiction of cylinder movements/transfers).

1.1.5.5.2 Feed Operations

Feed operations are performed in the Feed Area of the X-3346 building. See Figure 1.1.5.5.2-1 (located in Appendix E) for a function depiction of the feed process. The feed system is designed to supply UF_6 to the enrichment process located in the X-3001 and X-3002 buildings. The feed system sublimes UF_6 from cylinders placed in electronically heated feed ovens. The feed system also is connected to equipment to increase the purity of the UF_6 fed to the enrichment process by removing non-UF_6 gases from the feed cylinder prior to feeding. UF_6 may be fed from any approved UF_6 cylinder. Once the UF_6 has been vaporized and purified, the UF_6 gas is transferred by desublimation into one of the six freezer/sublimers used for feed. When feed is needed for the Process Buildings, it is sublimed from the freezer/sublimer and is passed through the feed system pressure reducing station before it is fed to the enrichment process via the X-2232C Interconnecting Process Piping (IPP). The feed system can supply to two feed streams at two different feed rates to the enrichment process. Feed can also be provided to the IPP by bypassing the freezer/sublimers and feeding the pressure reducing station directly. The capability is also available to provide feed material to the Withdrawal Area so that it can be used to blend with product UF_6 from the freezer/sublimers. Feed from the feed manifold can be transferred to the dump cylinders in the Feed Area as can feed from four of the feed ovens.

Empty feed cylinders are staged on the X-7746S or X-7746W Cylinder Storage Yards prior to shipment from the X-3346A building. The source and customer cylinders are staged on the X-7746W or X-7746S Cylinder Storage Yards prior to sampling and shipment of the customer cylinders from the X-3346A building.

Feed ovens are the primary components in the feed process. Feed ovens are enclosures that restrict air-leakage to provide efficient heating of the cylinders, but are not designed as pressure vessels. The ovens heat the cylinders utilizing electrically heated air. UF_6 is sublimed from the solid phase into a vapor for enrichment in the process buildings. The feed process has several stages. The feed is vaporized, monitored for "lights," and fed to freezer sublimers to be purified (removal of lights) and desublimed. The feed is held in freezer/sublimers, vaporized (sublimed), and pressure controlled before entering the process buildings. "Lights" refer to light gases (e.g., N_2, O_2, HF, etc.) entrained in the feed material. There are two feed headers located in the Feed Area. The oven heating system is programmed to hold the air temperature constant such that the cylinder wall temperature is held at approximately 185° Fahrenheit (F). When the cylinder weight reaches a determined value, the temperature of the feed oven and the rate of feeding is decreased until the cylinder is nearly empty. Any solid UF_6 left in the feed cylinder after the feed rate declines to a predetermined level is "heeled" into the X-2232C feed piping downstream of the pressure reducing station until the cylinder pressure is equal to that of the X-2232C feed piping. "Heeling" is the process for removing residual UF_6 from a cylinder when it can no longer be used to feed material into the cascade. The emptied feed cylinder is then moved on to storage. See Figure 1.1.5.5.2-2 (located in Appendix E) for a typical depiction of a feed oven.

1.1.5.5.3 Enrichment Operations

The enrichment process is contained in the X-3001 and X-3002 buildings. See Figures 1.1.5.5.3-1, 1.1.5.5.3-2, 1.1.5.5.3-3, 1.1.5.5.3-5, 1.1.5.5.3-6, 1.1.5.5.3-7 (located in Appendix E) and 1.1.5.5.3-4 (located in Appendix A) for a functional depiction of the enrichment process. Each process building contains multiple cascades to optimize operating costs and production flexibility. Each cascade is capable of enriching UF_6 gas to the desired product assay. UF_6 feed material is supplied from the X-3346 building Feed Area to the process buildings via the X-2232C IPP. In the process buildings, feed is distributed to the feed control systems for each cascade. The feed flow rates to each cascade are automatically controlled to ensure the desired feed is added to the cascade to support the production rate. As the feed enters the cascade, it is mixed with material already in the cascade and is separated into enriched and depleted material streams. This process continues until the material exits the top of the cascade as enriched product or the bottom of the cascade as tails material. The proportion of feed that becomes enriched product is controlled by the stage control valves, which are adjusted to provide the desired product and tails assays. Product and tails material are withdrawn from each cascade and sent to the X-3346 building Withdrawal Area via the X-2232C piping for transfer to cylinders. The product is sublimed directly into product cylinders through vacuum pump transfer. The tails material is sublimed directly into tails cylinders through compressor transfer. The commercial ACP cascade is limited to a maximum assay of 10 wt. percent ^{235}U.

The major components that support the enrichment operations are: centrifuges; centrifuge floor mount systems; service modules; inter-machine flow and control; X-2232C piping; and isolations valves.

1.1.5.5.3.1 Centrifuges

The gas centrifuge is comprised of a number of subassemblies (see Figure 1.1-12): Casing; Rotor; Column; Upper Suspension Assembly (USA); Lower Suspension and Drive Assembly (LSDA); and the Molecular Pump. A more extensive description of each of these components can be found in the ISA Summary.

1.1.5.5.3.2 Floor Mount

The centrifuge mount system is the primary structural interface between the soil subgrade of the process building floors and the centrifuges. The centrifuge mount system is a hard-torsion, hard-shear, and soft-rocking system. It consists of recessed steel floor modules encased in a large isolated concrete foundation mat. A mount at the bottom of the floor module, known as the fifth point, is designed to carry the full vertical weight of the centrifuge. Four specialty designed anchor pins with elastomeric isolators are arranged in a symmetrical pattern around the base of each centrifuge at the operating floor level. These pins attach the centrifuge to the encased steel frame and provide hard shear resistance in the event of horizontal thrust or torque lock-up, but allow vertical movement at the pin for the rocking motion.

The centrifuge mount system is designed so that each centrifuge responds to its operating environment independently of other centrifuges. This is accomplished by having the massive concrete foundation mitigate the effects of torque and shear experienced during an operational upset such as a rotor failure. The overturning forces experienced during an operational upset or by external events such as an earthquake are attenuated by the centrifuge mount's soft rocking suspension.

1.1.5.5.3.3 Service Module

The piping configuration used to connect the centrifuges in the UF_6 enrichment process is designed to minimize the likelihood of a major interruption of operations, provide isolation of centrifuges and minimize construction costs. 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, miss-operation of the UF_6 feed system, and centrifuges that are encountering operational problems. Figure 1.1-14 (located in Appendix B) depicts the Service Module and its general layout and systems interfaces.

Within the process building, utilities and process piping are routed to the centrifuges via service modules that consist of a 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, vacuum, and recycle are typically stainless steel, while those for air, cooling water, and fire suppression are steel. Smaller branch pipes connect the headers to each of the centrifuges. The centrifuge isolation valves, centrifuge power controls, and centrifuge instrumentation are also mounted on the service modules. Each service module services multiple centrifuges and the service modules are connected in series to support an operating cascade.

1.1.5.5.3.4 Inter-Machine Flow and Control

The inter-machine flow and control system consists of process piping headers and valves for transporting the process gas to and from the centrifuges; feed control system for controlling the feed rate to the cascades in each train; inventory control system for each stage, which maintains the proper backpressure on each stage; instrumentation and controls for header pressures and centrifuge status; and sampling taps to provide sampling capability to determine product and tails assays and product contaminants.

1.1.5.5.4 Withdrawal Operations

Product withdrawal occurs in the Withdrawal Area of the X-3346 Feed and Withdrawal via desublimation directly into cylinders inside cold boxes. As many as four product assays can be fed to the X-3346 building from four separate dedicated half-building product lines from the process buildings. UF_6 can also be fed to the X-3346 Withdrawal Area from the X-3346 Feed Area for use as blend material to meet customer specifications. See Figure 1.1.3-1 (located in Appendix E) for a functional depiction of the product withdrawal process. Product material is first transferred through a series of vacuum pumps (vacuum pump trains) connected to the product line in the X-2232C piping and then desublimed directly into selected source or product cylinders which are located in cold boxes and does not involve UF_6 pressures above atmospheric pressure. Connection and disconnection of the couplings to the cylinders is supported by the Evacuation System in the Withdrawal Area of the X-3346 building which draws effluent through evacuation cold traps and chemical traps before venting through a permitted vent. The cold traps are heated and the UF_6 is desublimed into one of two dump cylinders located in cold boxes. The filled source or product cylinders are then moved to interim storage and can subsequently be moved to the X-3344 building for sampling and/or moved to the blending/transfer area in the X-3346 Feed Area. Interim storage can be in the X-3346 building or the X-7746W or X-7746S Cylinder Storage Yards.

Tails withdrawal, also in the Withdrawal Area of the X-3346 Feed and Withdrawal Building, is accomplished through compression and direct desublimation of UF_6 material into tails cylinders inside a cold box and does not involve UF_6 pressures above atmospheric pressure. The tails withdrawal design incorporates the capability for simultaneously withdrawing two uranium assays. The compression train consists of centrifugal compressors arranged in series with coolers and with recycle capability. Tails withdrawal is used for emergency inventory removal. See Figure 1.1.5.5.4-1 (located in Appendix E) for a functional depiction of the tails withdrawal process. Effluent protection for cylinder connection and disconnection is the same as for product cylinders.

The major components that support the withdrawal operations are vacuum pump trains, tails, withdrawal trains, cold boxes, cold traps, chemical traps, assay spectrometers, and vents. See

Figures 1.1.5.5.4-2 and 1.1.3.3.1-4 (located in Appendix E) for a typical depiction of a tails compressor and a cold box. See Figure 1.1.3-1 (located in Appendix E) for a depiction of the vent system.

1.1.5.5.5 Sampling Operations

UF_6 sampling operations for UF_6 product material is carried out in the X-3344 building, also known as the Customer Services Building. See Figure 1.1.5.5.5-1 (located in Appendix E) for a functional depiction of the sampling process. American Society for Testing and Materials (ASTM) sampling standards necessitate that sampling must be from homogenized UF_6; the design involves liquefaction of UF_6 during sampling operations (Reference 19 and 20). In addition, some sampling of feed and tails cylinders is done to support Nuclear Material Control and Accountability requirements.

Autoclaves with heating and cooling capability are used to liquefy UF_6 in the cylinders to facilitate sampling and then solidification of the UF_6 in the cylinders at the end of the sampling. A cylinder may be any approved UF_6 cylinder per ANSI N14.1 (Reference 24) that meets nuclear criticality safety (NCS) requirements. The autoclaves are pressure vessels and are designed to contain a UF_6 release. Electrically heated hot air is the heating medium and cold air is used for cooling.

The major components that comprise the sampling and transfer operations are autoclaves, cold traps, and vents. See Figure 1.1.3.3.3-2 (located in Appendix E) for a typical depiction of an autoclave. See Figure 1.1.5.5.5-2 (located in Appendix E) for a functional depiction of the vent system.

1.1.5.5.6 Blending/Transfer Operations

Blending/transfer operations may be performed in the Feed Area of the X-3346. Blending is performed if the assay of enriched UF_6 needs to be adjusted to meet customer specifications. Transfer between cylinders is performed if the assay of the UF_6 meets customer specifications. A capability is also available to provide feed material from the Feed Area to the Withdrawal Area so that it can be used to blend with product UF_6 as it is being withdrawn through four separated product pipes.

Localized blending of enriched UF_6 between cylinders and/or gaseous transfer of enriched UF_6 between cylinders is performed using a combination of up to three dedicated feed ovens and five dedicated cold boxes. Blending is performed by sublimation transfer of the UF_6 from parent cylinders (uranium feed cylinders and source cylinders) to a daughter cylinder by desublimation to meet customer specifications normally in a customer cylinder. The parent cylinders are heated in the feed ovens to sublime the UF_6 and the UF_6 is then desublimed directly into a daughter cylinder in a cold box. The transfer of enriched UF_6 from a parent source cylinder directly into customer cylinders may also be done using a dedicated feed oven and cold box in the same fashion. Transfer/blending does not involve UF_6 pressures above atmospheric pressure.

The major components that comprise the blending/transfer operations are feed ovens, cold boxes, cold traps, and vents. See Figure 1.1.3-1 (located in Appendix E) for a functional depiction of the vent system.

1.1.5.5.7 Shipping Operations

The X-3346A building is also the shipping point for emptied cylinders leaving the ACP as well as UF_6 cylinders shipped to fulfill customer product orders (including Russian LEU), and UF_6 cylinders containing feed or depleted material. Any approved UF_6 cylinder may be shipped from this facility. See Figure 1.1-4 (located in Appendix B) for a schematic of the Feed, Withdrawal, and Product Operations.

Filled customer product cylinders, emptied feed cylinders, and other UF_6 cylinders will be prepared for shipment and shipped in accordance with U.S. Nuclear Regulatory Commission (NRC) and DOT regulatory requirements from the X-3346A.

1.1.5.5.8 Waste Handling Operations

Depleted UF_6 tails material is considered a resource material with the ultimate disposition to be determined and is not considered a waste. The Licensee intends to evaluate possible commercial uses for depleted UF_6. Depleted UF_6 is stored in steel cylinders within cylinder storage yards until this material can be processed in accordance with the disposition strategy established by the Licensee. Depending upon technological developments and the existence of facilities available prior to the ACP shutdown, the depleted UF_6 may have commercial value and may be marketable for further enrichment or other processes.

Waste generated by the ACP is collected, handled, packaged, segregated, stored, and shipped for off-site treatment/disposal in a safe and environmentally acceptable manner in accordance with applicable state and federal regulations, and plant procedures. Waste accumulation areas are established throughout the ACP as necessary to meet these regulatory requirements.

The ACP obtains waste management services from a qualified provider licensed/certified by the NRC or an agreement state. Waste may be further sampled/measured to assist in determining the proper waste characterization and proper disposal/treatment method.

Potential waste streams generated include Low-Level Radioactive Waste, LLMW, RCRA Hazardous Waste, Sanitary/Industrial Waste, Recyclable Waste, and Classified/Sensitive Waste.

Waste generating activities are evaluated for waste minimization opportunities to reduce the volume and toxicity of waste generated to the degree determined to be economically practicable.

A further description of the transportation impacts can be found in Section 4.2 and the waste impacts can be found in Section 4.13 of the Environmental Report for the American Centrifuge Plant.

1.1.5.5.9 Liquid and Air Waste Discharge Points

Waste discharge points are categorized by either liquid (water) or air.

For liquid, wastewater discharges are handled by different means depending upon the originating source: process, sanitary, or storm water.

No process wastewater is intentionally discharged from the liquid effluent tanks. Accumulated water in these tanks are sampled and managed according to analytical results. Trained professionals using approved spill response protocols and spill response equipment will promptly contain liquid spills within the process buildings. Spill materials will be collected, sampled, analyzed, and managed in accordance with applicable federal and state laws. The only intentional process wastewater discharge resulting from plant operations is the blow down from the TWC (Tower Cooling Water) system. This cooling water system is not interconnected with the MCW (Machine Cooling Water) system located in the process buildings. The MCW system is a closed-loop system, which requires minimal makeup water, but does not have blow down discharges.

Sanitary wastewater (e.g., showers, toilets, etc.) located within the area discharge to the plant sanitary sewer system and ultimately to the X-6619 Sewage Treatment Plant. Treated sanitary wastewaters are discharged from X-6619 directly to the Scioto River via an underground pipeline via a permitted NPDES outfall.

Storm water runoff from the ACP area, along with some once-through cooling water (sanitary water), drain to a pair of holding ponds (X-2230N West Holding Pond and X-2230M Southwest Holding Pond). These ponds provide a quiescent zone for settling suspended solids, dissipation of chlorine, and oil diversion and containment. The ponds discharge to unnamed tributaries of the Scioto River. An automated sampler collects a weekly composite sample of the liquid effluent for radiological analysis as well as NPDES-mandated analyses.

For air, the process release of hazardous gases to the atmosphere is the area of concern. The projected concentration of Hydrofluoric acid (HF) gas release is six orders of magnitude, or a million times less than the Threshold Limiting Value (TLV) for HF. The conservative estimates of HF concentrations at the DOE reservation boundary indicate that its release during ACP operations will have an insignificant impact on air quality. On the other side, each process area vent systems in the X-3001, X-3002, X-3344, X-3346, and X-7725 buildings have gas flow monitoring instrumentation with local readouts as well as analytical instrumentation to continuously sample, monitor, and to alarm if UF_6 should breakthrough in the effluent gas stream.

1.1.5.6 Enrichment Process Support Systems

Support systems that support the enrichment process include the Area Control Room (ACR), vacuum systems (i.e., Evacuation Vacuum [EV] and Purge Vacuum [PV]), Machine Cooling Water, Criticality Accident Alarm System (CAAS), portable gulpers, and building HVAC systems.

1.1.5.6.1 Control Centers

There are two Area Control Rooms (ACRs) that support the ACP. One ACR is located in the X-3012 building and supports the enrichment process in the X-3001 and X-3002 buildings. X-3346 building has an ACR that supports the feed, blending/transfer and withdrawal operations performed in the X-3346 building and the sampling operations performed in the X-3344 building.

The Local Control Centers (LCC) are located in the process area and are designed to control a portion of a process building equipment. The LCCs are connected to the ACR that is designed to control an entire process building. The process may be controlled at the appropriate LCC or ACR. This will include monitoring of centrifuge parameters, service module header pressures, process gas pressures, building temperatures, and operation of the Intermediate Flow and Control System, as well as information about the EV and PV systems. The Intermediate Flow and Control System consist of four subsystems: 1) process piping headers; 2) feed control system; 3) inventory control system; and 4) controls.

The X-3012 building houses the ACR for the X-3001 and X-3002 buildings. The ACR is designed to control the centrifuges in both process buildings. The ACR, along with the LCCs, are used to monitor and control the centrifuges and cascade parameters. Each centrifuge has operating parameters that are monitored to measure the centrifuge condition and operating efficiency. Operations personnel investigate deviations from normal operating conditions and adjustments to the centrifuge are made to correct any problems.

The X-3346 building has an ACR for housing the monitoring, control, and alarm equipment associated with the feed, blending/transfer, withdrawal operations in the X-3346 and the sampling operations in the X-3344 building. This includes the assay spectrometers for monitoring feed, product and tails.

The ACR computer system displays an overview of the process equipment and utilities in process buildings. From the ACR, the operators can monitor utilities, and process variables in the cascade and centrifuge level. Also, operators can change setpoints (within certain parameters), isolate parts of the process, receive and identify alarm sources, and dispatch service personnel.

The status of each process controller can be displayed. A change in status activates an alarm. In the event of failure of a process controller, a standby controller automatically takes control of the system. The controllers interface directly with process equipment. Under normal circumstances, the LCCs are unmanned. However, in case of a failure, the LCCs can be used to provide the operators with the capability to control the appropriate equipment.

1.1.5.6.2 Vacuum Systems

To mitigate and prevent degradation or failure of key centrifuge components, the centrifuges operate in a vacuum environment. There are two major vacuum systems: EV and PV Systems (see Figure 1.1-15). Each centrifuge is connected to both systems via a manual interlock, so that the centrifuge can only be connected to one system at a time. Each EV system includes two mechanical vacuum pumps, valves, and controls to permit a vacuum pump to serve as a spare for the other. The EV system also includes piping required to connect the centrifuges from the molecular pump through the service module piping to the mechanical vacuum pumps, and piping from the discharge of the mechanical headers. The EV system is used for roughing pump down of service module headers and newly installed centrifuges. Each PV system includes two mechanical vacuum pumps, valves, and controls to permit a vacuum pump to serve as a spare for the other. The PV pumps discharge to a set of alumina traps to remove any trace quantities of UF_6 prior to the gases being vented to atmosphere. The PV system also includes piping required to connect the centrifuges from the molecular pump through the service module piping to the mechanical vacuum pumps, and piping from the discharge of the mechanical headers. The PV system is used as a final pump down of installed centrifuges, and to maintain a continuous vacuum source on the centrifuge, when it is in operation. See Figures 1.1.5.5.8-1 and 1.1.5.5.8-2 (located in Appendix E) for a functional depiction of the EV/PV system.

1.1.5.6.3 Machine Cooling Water System

The Machine Cooling Water (MCW) system is a closed-loop circulating water system designed to provide continuous cooling of the centrifuge LSDAs, and the PV, and EV pumps. The system contains circulating water pumps, filters, heat exchangers, expansion tanks, and piping tie-ins 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 water pumps. The tank provides a convenient point for adding make-up water and water treatment chemicals. The discharge of the circulating pumps passes through a MCW filter and a heat exchanger where the MCW is cooled. The heat exchanger cooling water is supplied from a closed-loop Chilled Water (CW) system and the CW chiller (heat exchanger) cooling water is supplied from the cooling tower and Tower Water Cooling (TWC) pumps. The cooled MCW then returns to the centrifuges 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 where chemicals are added via a chemical injection pump.

Sanitary water is provided for the MCW make-up water and the chilled water closed-loop. This water passes through a deionizer before entering either the MCW closed-loop or chilled water closed-loop. The make-up water is used for initial fill purposes and for maintaining the proper level of MCW and CW in the system. MCW system alarms are monitored in the ACR.

1.1.5.6.4 Building Heating, Ventilation, and Air Conditioning Systems

Process building heating, ventilation, and air conditioning (HVAC) systems are designed to maintain the building environment required for proper operation of process and associated equipment. The main subsystems affecting process buildings are the Process Area Ventilation System, and Process Area Heating and Pressurization System.

The Process Area Ventilation System provides air circulation and, when necessary, cooling using outside air. Each ventilation subsystem consists of a supply fan, return/exhaust fan, filters, and associated ductwork with automatic dampers and controls. The return/exhaust air fan draws heated air from the centrifuge 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 area served by the subsystem, some percentage of outside air, up to 100 percent, is drawn through a damper into the supply fan plenum. This outside air mixes with any return air 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 of the service module structure. Air is directed downward from the service module duct. No heating coils are utilized in this system.

The Process Area Heating and Pressurization System heats outside make-up air and supplies enough heat to offset exterior wall and roof heat losses. This system also serves to maintain a positive indoor pressure relative to the outdoor pressure. Individual heating and pressurization units are located on the mezzanine in the process buildings. Each unit consists of pneumatically operated outside air intake damper, a return air damper, a filter section, a heating coil (face and bypass) section, a supply fan, and distribution ducts that form a perimeter boundary around the centrifuge area. Outside air and return air dampers are modulated to maintain a positive building pressure. Recirculating Heating Water is supplied to the heating coils.

HVAC is provided to the X-3012, X-3344, X-3346, X-3346A, X-7725, and X-7726 buildings/facilities to provide proper operation of the equipment, as well as comfortable working conditions for personnel.

Other areas of the ACP are provided with HVAC or only heating and ventilation, depending on the location and function of the area or facility. Supplemental heat can be provided in any ACP facility using portable electric heaters should the RHW be out of service or outside weather conditions dictate the need.

1.1.5.6.5 Criticality Accident Alarm System

The primary radiation alarm system is the CAAS designed to detect a nuclear criticality and provide audible and visual alarms that will alert personnel to evacuate the immediate area. ACP primary facilities that handle ^{235}U in quantities exceeding 700g and enrichment levels greater than or equal to 1 weight percent have CAAS coverage except the UF_6 cylinder storage yards. An exemption for the UF_6 cylinder storage yards has been requested in Section 1.2.5 of this License Application. Cylinders are moved between the various buildings with the material in a solid state on approved and defined routes using specifically designed equipment in accordance with approved procedures that are covered by CAAS.

Operations involving fissile material are evaluated for Nuclear Criticality Safety (NCS) considerations prior to initiation. The need for CAAS coverage is considered during the evaluation process. Coverage is provided, unless it is determined that coverage is not required and the finding is documented in a NCS Evaluation. Per 10 CFR 70.24, CAAS is required in each area where threshold quantities (e.g., more than 700 grams of ^{235}U) of special nuclear material are handled, used, or stored. The CAAS coverage areas are identified on plant drawings, and controls are established to preclude special nuclear material from areas where coverage is not provided.

1.1.5.6.6 Portable Gulpers

A portable gulper system is used for localized exhaust on applications like small-scale maintenance tasks. The gulper inlet duct or hose is placed near the work area. Any escaping airborne contamination is removed from the source and passes through the duct or hose and into the filter bank, where, depending on the operation, gases are neutralized and the particulates are removed. The resultant exhaust is clean air that is typically discharged into the work area.

1.1.5.7 Centrifuge Assembly and Movement Systems

1.1.5.7.1 Centrifuge Assembly

The centrifuges are assembled in the X-7725 building and/or the X-7726 facility assembly stands. Parts for the centrifuge assembly are received at these locations. Secure facilities are available to receive and store the classified parts, as well as other components of the centrifuges. Overhead cranes, fork trucks, and parts elevators are available to handle parts delivery to the assembly stands.

Two centrifuge assembly positions and a column assembly stand is provided in the X-7726 facility and up to six centrifuge assembly positions and six-column assembly stands are available in X-7725 building for assembly of the various components into a completed centrifuge. 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 required during the assembly of the centrifuges. Gross leak testing may be performed at these locations before the assembled centrifuge is moved from the assembly stands. No process gas (UF_6) testing of the centrifuges will take place in the assembly areas. Completed centrifuges 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 buildings. Testing of the centrifuges using UF_6 may be performed in the X-7725 building Gas Test Stands or in the process buildings after installation, prior to being placed into service.

1.1.5.7.2 Centrifuge Transporter Cart

The centrifuge transport system, consisting of the centrifuge transporter cart and the various building crane systems, is used to move centrifuges. Centrifuges are transported between the X-7725 building and X-7726 facility assembly facilities and the X-3001 and X-3002 buildings within the X-7727H corridor using a centrifuge transporter cart. Within a building, centrifuges are moved using overhead cranes from assembly locations to storage locations, or between the storage locations and the centrifuge transporter cart.

The centrifuge transporter cart is a battery-operated, mobile vehicle specially designed to transport centrifuges in an upright position, while protecting them from damage due to excessive motion. The centrifuge transporter cart includes a tugger vehicle and can accommodate a maximum of two centrifuges. The centrifuge transporter cart is equipped with mechanisms to secure each centrifuge in a vertical position during the different modes of operation. The design assures that the centrifuge transporter cart remains stable and level during loading and unloading operations.

1.1.5.7.3 Cranes

There are a variety of cranes that will be used. Depending on the operation they support, they will vary in configuration, span length, and capacity. Some cranes will be for general use, whereas others are designed for specific tasks and applications. Crane designs are in accordance with recognized national standards such as the American Society of Mechanical Engineering (ASME)/American National Standards Institute (ANSI) B30 series, the National Electric Code, and the Crane Manufacturing Association of America. There are numerous specialty cranes and monorails located throughout the ACP that support specific operations.

There are specialty cranes in the process buildings for installing and removing centrifuges. Crane features include variable speed controls, strict deflection criteria, clamping devices for centrifuge movement, and automated positioning controls.

The crane systems in X-7725 building and X-7726 facilities were specifically designed for receiving, assembly and disassembly of the centrifuges. The X-7725 building features a sophisticated under hung crane system on the main and upper assembly levels. Operator controlled cabs are able to transfer between adjoining remote controlled bridges providing mobility throughout the assembly area.

The feed, withdrawal and sampling operations feature cranes for movement of cylinders to and from exterior storage lots. Except for the X-3346 Feed Area, the cranes do not enter the buildings. The cranes are operated from the ground by pendant or by remote control and are specifically designed for handling cylinders.

1.1.5.7.4 Cylinder Transporter

The cylinder transporters used in the X-3346 Feed Area is a rail mounted transporter that is loaded by a bridge crane internal to the X-3346 building. The cylinder transporter is designed to support weighing the cylinder and cylinder cradle. The transporter is designed to move the cylinder and cradle to the designated feed oven and onto the cylinder carriage system. The cylinder transporter is also designed to remove the cylinder from the feed oven cylinder carriage system and to place the cylinder and cradle on accountability scales for measurements required by the NMC&A Program. The cylinder and cradle are removed from the cylinder transporter by a bridge crane internal to the X-3346 building. The cylinder transporters used for the X-3346 Withdrawal Area and in the X-3344 Customer Services Building function in the same fashion as described above in loading and unloading the cylinder and cradle into the cold boxes and autoclaves respectively. Cranes place the cylinders and cradles on the cylinder transporters externally to these two areas.

The cylinder transporter is electrically powered from rechargeable batteries on the transporter. The cylinder transporter is designed to be locally controlled.

1.1.5.8 Plant Support Systems

Plant support systems consist of the following: electrical distribution system (345 kV, 13.8 kV, 4,160 volt [V], 2,400V, 480V, 277V, 208V, and 120V); instrument air; TWC; fire and sanitary water storage and distribution systems; and sewage treatment system.

1.1.6 Hazardous Material Storage

Large quantities of highly hazardous material, defined as a Threshold Quantity (TQ) in the Occupational Safety and Health Administration (OSHA) Process Safety Management Standard (29 CFR 1910.119) and the EPA Risk Management Program Standard (40 CFR Part 68), are not present in the ACP.

Other chemicals and typical industrial materials (e.g., acetone, solvents, acids and oils) are used in the X-7725 building, X-7726 facility, and X-3012 building for assembly and maintenance activities. These substances are stored in approved containers and are listed in the Hazardous Material Inventory Control System. Quantities are appropriately reported annually to the Federal and State EPA as required by the *Superfund Amendments Reauthorization Act* (SARA Sections 312 and 313).

The Licensee complies with requirements for generators of hazardous and mixed waste. The State of Ohio has adopted a federal conditional exemption from the hazardous waste rules that is available under 40 CFR Part 266, Subpart N (OAC 3745-266).

1.1.7 Roadways

Two major four-lane highways service the DOE reservation: U.S. Route 23, traversing north-south, and State Route 32/124, traversing east-west. The reservation is situated approximately three and one half miles from the intersection of U.S. Route 23 and State Route 32/124. 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 fenced portion of the DOE reservation. Alternative ingress and egress from the reservation can be established from the north access road in the event of significant Main Access Road repairs. Service roads throughout the reservation connect to the Perimeter Road with access to the ACP controlled through security portals. The reservation roadways are depicted in Figures 1.1-1 and 1.1-2 (located in Appendix B).

1.1.8 Phased Modular Expansion Plan for the American Centrifuge Plant

It is the intent of ACO to deploy portions of the ACP in a modular fashion to accommodate market demand on a scalable, economical gradation. This modular deployment may encompass utilization of cascades of Low Enriched Uranium (LEU) production for LEU customer product or feed material into HALEU cascades. The ratio of LEU cascades to HALEU cascades would be approximately 6 to 1.

1.1.8.1 High Assay Low Enriched Uranium Demonstration

The HALEU Demonstration cascade utilizes a similar centrifuge design to that used for the Lead Cascade. The equipment necessary to perform the enrichment process is in the X-3001 Process Building and consists of product and tails withdrawal system, UF_6 cylinders, centrifuges, and supporting systems. The product and tails withdrawal systems use three cold boxes. NaF traps are used for additional withdrawal capacity during dumping. A 30B UF_6 cylinder is used for the feed material. Centrifuges and supporting units are placed in the Train 3 area of the X-3001 building. For further plant and process specifics related to the HALEU Demonstration Program, refer to LA-3605-0003A, *Addendum 1 of the Integrated Safety Analysis for the American Centrifuge Plant – HALEU Demonstration* (Reference 7).

In support of this HALEU Demonstration Program and NRC Materials License (SNM-2011) Condition 23, DOE amended the *Appendix 1 Lease Agreement between the U.S. Department of Energy and United States Enrichment Corporation for the Gas Centrifuge Enrichment Plant* (GCEP Lease Agreement) (Reference 71). The amended GCEP Lease Agreement renewed and extended the term of the lease through May 31, 2022. The ACO sublease incorporates the terms of the GCEP Lease Agreement.

At the conclusion of the three-year HALEU Demonstration Program, the facilities will be either returned to the DOE in accordance with the requirements of the GCEP Lease Agreement or the parties will amend the GCEP Lease Agreement to allow the performance of other work on the leased premises.

1.1.8.2 High Assay Low Enriched Uranium Demonstration Continuation

As the second phase of deployment, the Licensee plans to continue operation of the 16 centrifuge HALEU cascade as previously described for an additional 10-year period. The Licensee would amend the License Application and applicable Supporting Documents to allow continued operation of this HALEU cascade with increased possession limits for the requested extended period of operation. ACO's financial assurance and decommissioning liability would be established in accordance with the requirements of 10 CFR 70.38, 40.42, and 30.36 and submitted as part of the License Amendment Request.

This phase would only occur if parties agree to extend the GCEP Lease Agreement in support of ongoing planned Licensee activities. In accordance with Materials License Condition 23, the Licensee would provide a copy of the amended agreement to the NRC. Additionally, the Licensee would notify the NRC if/when a decision is made to transition to this phase seeking approval prior to the implementation of any changes.

1.1.8.3 High Assay Low Enriched Uranium Production

A subsequent proposed deployment will be the installation of one or more 120 centrifuge HALEU cascade(s) in Train 3 with HALEU Feed and Withdrawal stations located in Train 4.

1.1.8.4 Expanded Low Enriched Uranium and High Assay Low Enriched Uranium Production

The proposed follow on phase to High Assay Low Enriched Uranium production discussed in 1.1.8.3 above will be the addition of one or more 120 centrifuge HALEU cascades and/or LEU cascades and associated Feed and Withdrawal stations in a modular fashion all within the X-3001 building. The HALEU cascades could be fed directly from associated LEU cascades or directly with LEU cylinders.

1.1.8.5 Full ACP Deployment

The Licensee will notify the NRC in advance of the transition of the full ACP as previously approved with the initial issuance of Materials License SNM-2011. At that time, the Licensee will request a License Amendment and submit a detailed decommissioning cost estimate and required financial assurance documentation to NRC in accordance with the requirements of 10 CFR 70.38, 10 CFR 40.42, and 10 CFR 30.36 for NRC review and approval. Additionally, the Licensee will provide the necessary financial qualification documentation as detailed in Materials License Condition 15.

1.1.9 Material of Construction

The ACP facilities are designed and built in a manner to ensure an operating life of at least 30 years. Materials of construction are chosen in accordance with the guidance provided in GAT-901 and GAT-T-3000 (References 25 and 26) to ensure piping and other equipment can maintain a minimum wall thickness during the operating life of the ACP. Corrosion and erosion rates are not anticipated to exceed 0.0025 millimeter per year depending upon material of construction, equipment configurations and flow rates.

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

An example of the use of steel in this fashion is UF_6 cylinders. While steel will corrode and not produce a protective fluoride film, the design compensates for the corrosion by increasing the thickness of the cylinder wall. Operational requirements for periodic retesting of the cylinders every five years ensures that the residual wall thickness is still adequate even under high temperature conditions experienced during cylinder heating. Corrosion of steel is greatly increased if moisture is introduced into the UF_6 cylinders; however, controls are in place to minimize the presence of moisture to address criticality and chemical reaction concerns.

Soldering and brazing alloys must be considered for the effects of operational conditions, material compatibility, and corrosion over the expected life of the associated equipment to ensure the integrity of the equipment is maintained. These metals are also exposed to UF_6 and elevated temperature conditions which affect their corrosion rates. KY/L-1990 (Reference 27) is used as guidance in selecting soldering and brazing materials for process equipment. Experience from GDP operations with these materials of construction supports the expectation there should be no corrosion and erosion related breaches during the lifetime of the ACP because the design effort has considered the compatibility of materials, equipment, and process gas and its constituents.

1.1.10 Use of Lubricants

The ACP is designed and constructed to use oilless pumps and compressors as much as possible in the processing of UF_6. Where lubrication is required and the associated equipment can potentially see process gas, the preferred lubricants are compatible with UF_6 and HF. Compatible lubricants are polyfluoropolyethers (PFPE), known by shelf names such as Fomblin or Krytox. These lubricants are fluorinated which minimizes their ability to react with the fluorine associated with UF_6 and HF. The chemical components are carbon, fluorine, and oxygen. Also, PFPEs have minimal flammability and toxicity concerns.

When the process equipment cannot achieve the desired performance parameters utilizing fluorinated lubricants, hydrocarbon based lubricants can be used. Performance parameters include, but are not limited to, pressure, mass flow, and availability. Where hydrocarbon-based lubrication is required, the amounts in use are small enough such that criticality and combustible loading concerns are minimal.

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 U.S. Department of Energy Reservation in Piketon, Ohio

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 American Centrifuge Plant 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 X-3001 (X-3002) Typical General Equipment and Process Flow 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 1.1-4 Feed, Withdrawal, and Product Operations

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

Figure 1.1-5a X-3346 Feed Equipment and Process Flow 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 1.1-5b X-3346 Blending/Transfer Equipment and Process Flow

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

Figure 1.1-5c X-3346 Product Withdrawal Equipment and Process Flow

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

Figure 1.1-5d X-3346 Tails Withdrawal Equipment and Process Flow

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

Figure 1.1-5e X-3346 Typical General Equipment and Process Flow 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 1.1-6 X-3346A Typical General Equipment and Process Flow 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 1.1-7 X-3344 Typical General Equipment and Process Flow 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 1.1-8 X-7725 Typical General Equipment and Process Flow 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 1.1-9 X-7727H Typical General Equipment and Process Flow 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 1.1-10 X-2232C Typical General Equipment and Process Flow Layout

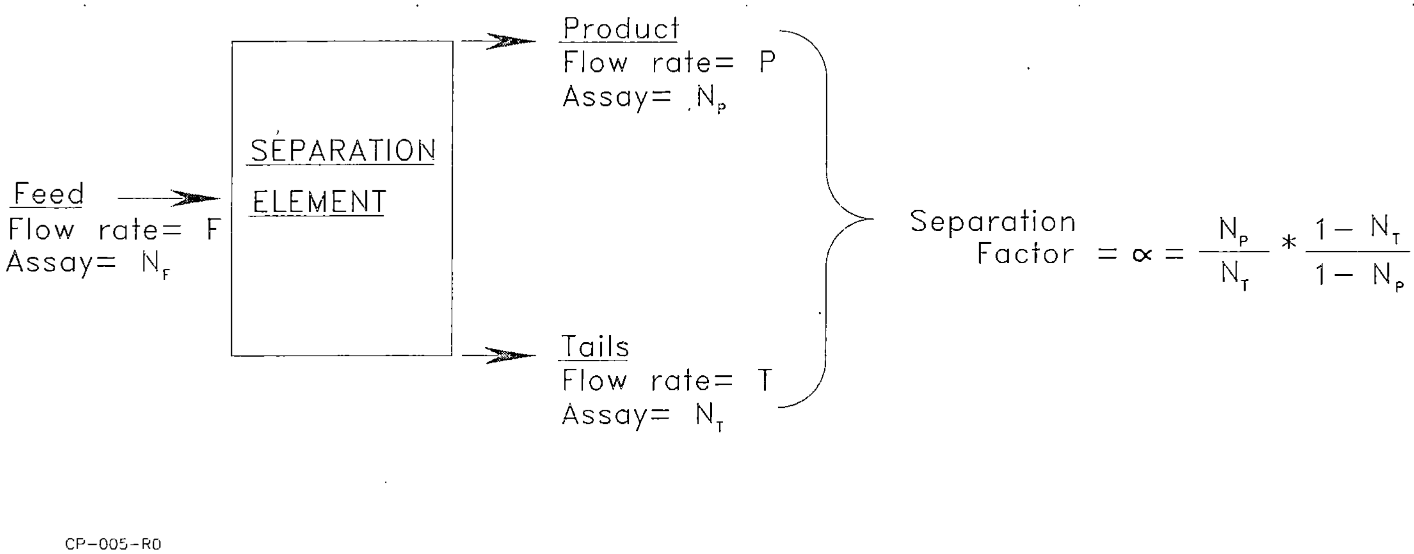

Figure 1.1-11 Separation Element

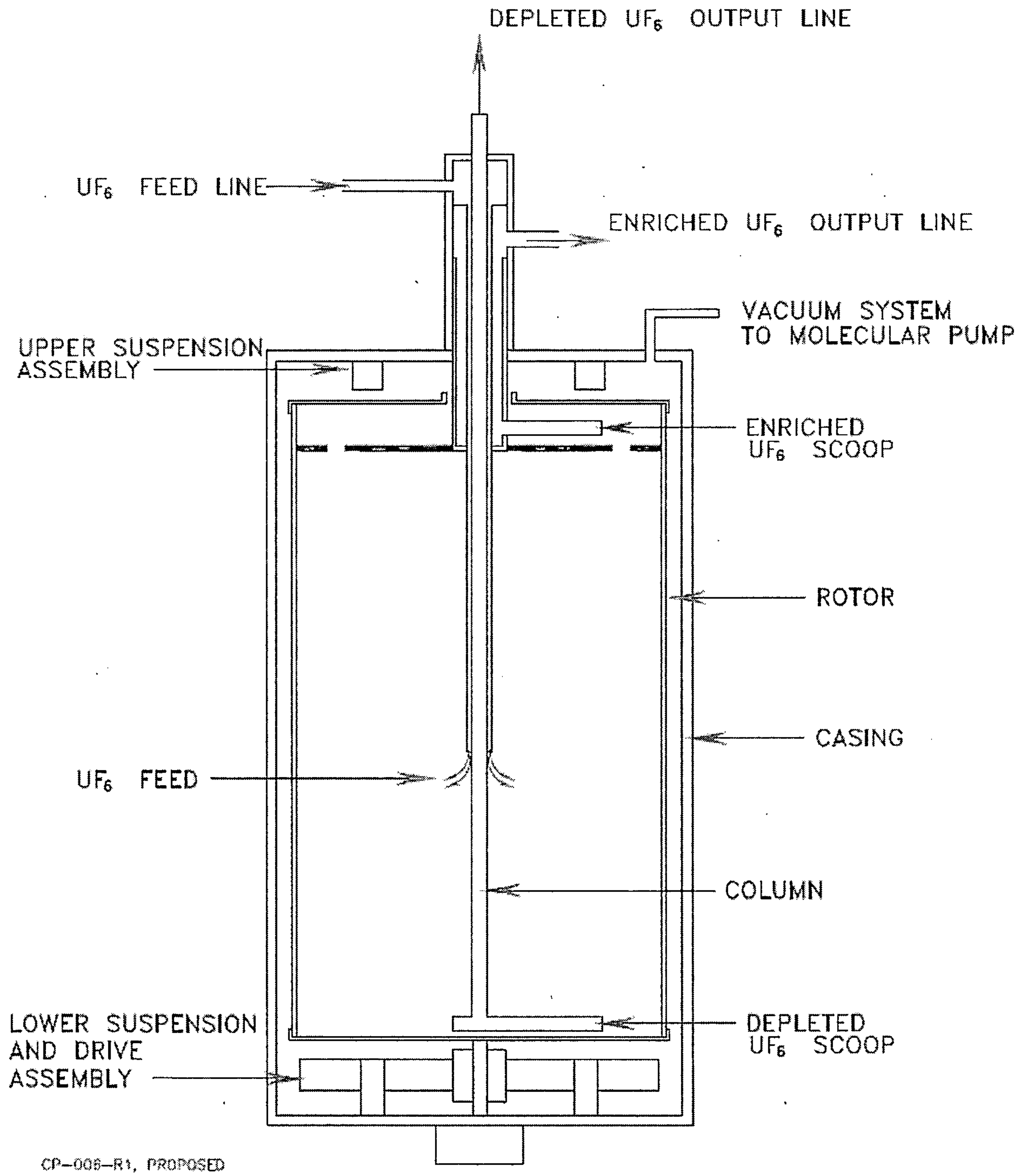

Figure 1.1-12 Centrifuge Schematic

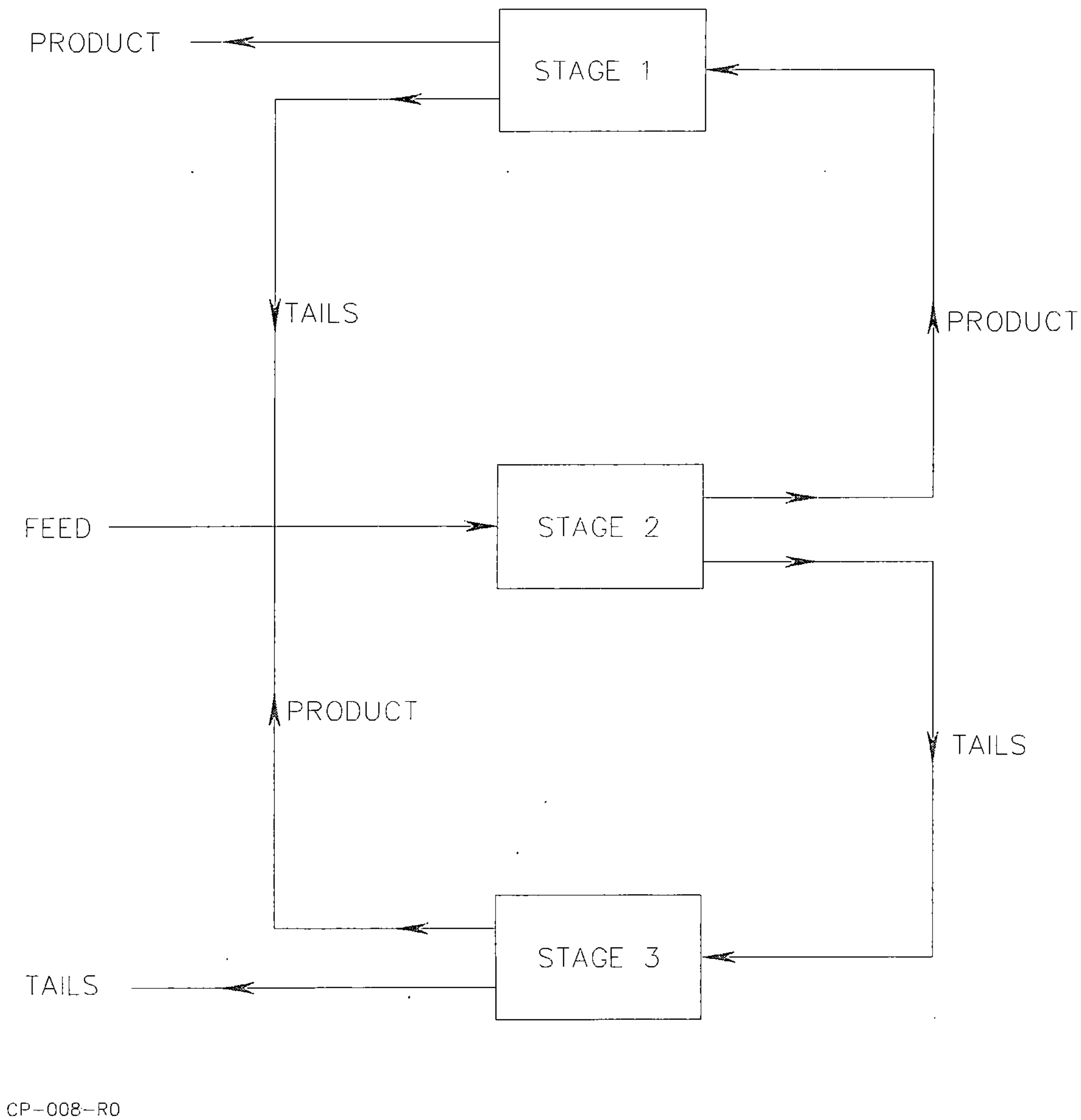

Figure 1.1-13 Example Cascade and Stage Flow Schematic

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

Figure 1.1-14 Systems Interfaces

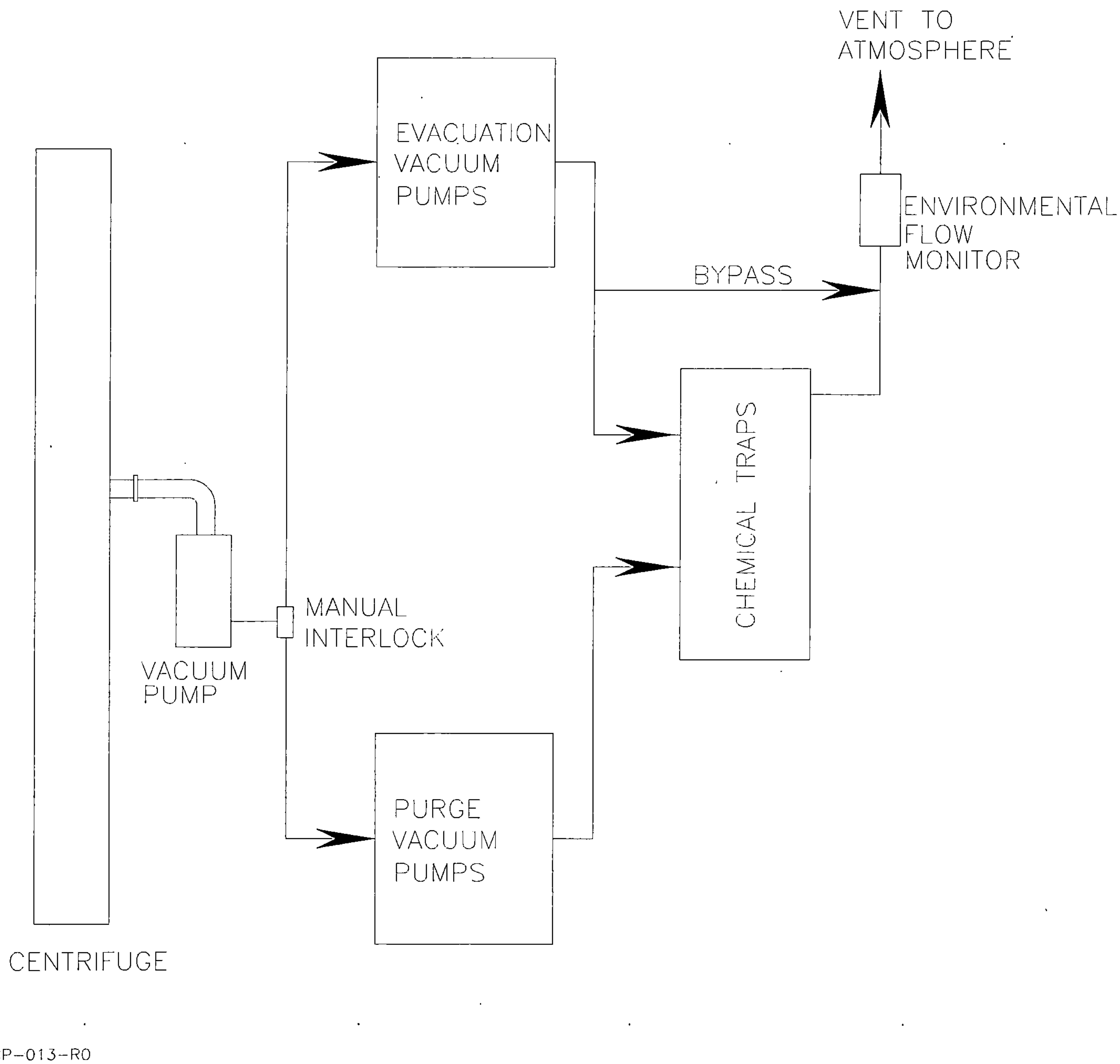

Figure 1.1-15 Purge and Evacuation Vacuum System Schematic

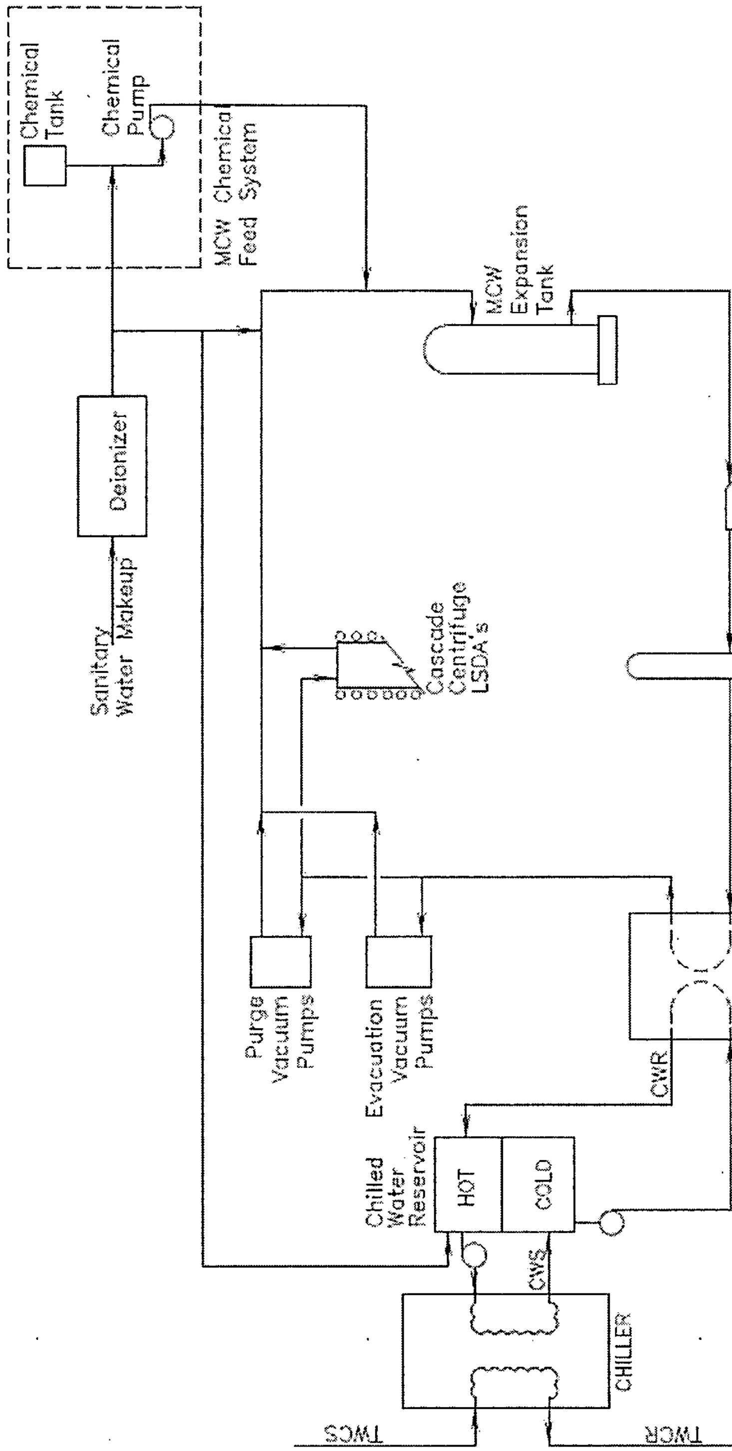

Figure 1.1-16 Machine Cooling Water System Flow Schematic

Table 1.1-1 American Centrifuge Plant Major Facilities

Facility No.	Facility Description	Facility Function
X-220E1	Evacuation Public Address System	Provides the ability to provide evacuation instructions or notification in the event of an incident requiring evacuation or sheltering of reservation/plant personnel.
X-220E3	Power Public Address System	Provides the ability to provide evacuation instructions or notification in the event of an incident requiring evacuation or sheltering of reservation/plant personnel.
X-220R	Public Warning Siren System	Provides notification to the public within a two-mile radius of the DOE reservation in the event of an incident requiring evacuation or sheltering of the public.
X-745G-2	Cylinder Storage Yard	Allows for movement and storage of UF_6 material outside of the process. (typically Tails).
X-745H	Cylinder Storage Yard	Future cylinder storage yard area reserved.
X-2202	Roads	Allow for easy and safe movement of people, equipment, and material.
X-2215A	Underground Electrical Distribution to Process Buildings	This facility provides 13.8 kV electrical power distribution to the process buildings.
X-2215B	Electrical Distribution to Areas Other Than Process Buildings	This facility provides 13.8 kV electrical power distribution to the process support facilities.
X-2220N	Security Access Control and Alarm System	Provides interior protection and high-security entry controls.
X-2230B	Sanitary Sewer	Provides underground sewage collection system.
X-2230C	Storm Sewer	Provides underground drainage system to collect surface water.
X-2230M	Southwest Holding Pond	Provide a quiescent zone for settling suspended solids, dissipation of chlorine, and oil diversion and containment prior to being discharged to an unnamed tributary of the Scioto River. Holding Pond #1
X-2230N	West Central Holding Pond	Provide a quiescent zone for settling suspended solids, dissipation of chlorine, and oil diversion and containment prior to being discharged to an unnamed tributary of the Scioto River. Holding Pond #2

Table 1.1-1 American Centrifuge Plant Major Facilities

Facility No.	Facility Description	Facility Function
X-2232C	Interconnecting Process Piping	Process piping that is external to the primary facilities that connects the X-3346 building to the X-3001 building and connects the X-3001 and X-3002 buildings (includes feed, product and tails UF_6).
X-3001	Process Building	Houses the centrifuges and their support systems.
X-3002	Process Building	Houses the centrifuges and their support systems.
X-3012	Process Support Building	Houses the operational and maintenance areas and the transfer aisleway that services the X-3002 building.
X-3344	Customer Services Building	Houses the equipment to sample cylinders for customer specifications as well as meeting NMC&A cylinder sampling requirements.
X-3346	Feed and Withdrawal Building	Houses four distinct areas of operation: one to meet the UF_6 feed material needs of the enrichment process operation, one to blend/transfer UF_6 between cylinders and two to meet the process withdrawal requirements: one for product withdrawal and the other for tails withdrawal.
X-3346A	Feed and Product Shipping and Receiving Building	Houses equipment necessary to receive and ship the UF_6 cylinders necessary to support the ACP operations as well as providing NMC&A scale capability.
X-5000	Switch House	This facility contains equipment necessary to distribute electrical power throughout ACP.
X-5001	Substation	This facility contains power transformers and other equipment necessary to transform 345 kV power to 13.8 kV for electrical power distribution throughout ACP.
X-5015	345 kV Underground Cable	This facility provides 345 kV electrical power from the X-530A to the X-5001.
X-6000	Cooling Tower Pump House, Air Plant, and Air Plant Support Systems	Contains the necessary equipment/systems to distribute dry compressed air to the ACP and to provide the requisite water to the X-6001 Cooling Tower for the removal of heat from the process buildings.

Table 1.1-1 American Centrifuge Plant Major Facilities

Facility No.	Facility Description	Facility Function
X-6001	Cooling Tower	Provides the necessary cooling requirements for the process buildings.
X-6002	Boiler System	Provides hot water for heating.
X-7721	Maintenance, Stores and Training Building	Provide areas for maintenance shops; stores and receiving activities; and training.
X-7725	Recycle/Assembly Building	An area where the centrifuges can be manufactured, assembled, tested, and maintained. Used as a shipping, receiving, and materials storage area.
X-7725A	Waste Accountability Facility	Serves as a storage area for equipment and parts necessary for the maintenance and repair of the process and process support equipment.
X-7725C	Chemical Storage Building	Provides clean, non-contaminated, protected, storage area of manufacturing chemicals.
X-7726	Centrifuge Training and Test Facility	Initially used for centrifuge component manufacturing and centrifuge assembly, then used for centrifuge assembly training and centrifuge component preparation.
X-7727H	Interplant Transfer Corridor	Provides a protected pathway to transport centrifuges from the X-7725 building or X-7726 facility to the process buildings or back, as necessary. This area also serves as a shipping and receiving area for equipment and components during construction.
X-7745R	Recycle/Assembly Storage Yard	Provides clean, non-contaminated, outside, horizontal rack storage of centrifuge casings prior to being moved inside the building for centrifuge assembly.
X-7746S	Cylinder Storage Yard	Allows for movement and storage of UF_6 material outside of the process.
X-7746W	Cylinder Storage Yard	Allows for movement and storage of UF_6 material outside of the process.

1.2 Institutional Information

ACO is the licensee for the ACP license to receive, acquire, possess, and transfer byproduct, source, and special nuclear material. ACO is a wholly owned indirect subsidiary of Centrus Energy Corp. (Centrus).

1.2.1 Corporate Identity

Centrus is a supplier of various components of nuclear fuel to utilities and advanced engineering, design, and manufacturing services to government and private sector customers. USEC Inc., the predecessor to Centrus, was organized in 1998 under Delaware law in connection with the privatization of the United States Enrichment Corporation. Centrus' direct and indirect subsidiaries are also registered companies in the State of Delaware.

Centrus' principal office is located at 6901 Rockledge Drive, Bethesda, MD 20817. Centrus is listed on the NYSE American under the symbol LEU. Private and institutional investors own the outstanding shares of Centrus. The principal officers of Centrus are listed below and are citizens of the United States.

Daniel B. Poneman, President and Chief Executive Officer
Larry B. Cutlip, Sr. Vice President, Field Operations

The NRC has determined that Centrus is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

In September 2008, USEC Inc., the predecessor to Centrus, formed five wholly owned subsidiaries in the State of Delaware to carry out future commercial activities related to the American Centrifuge project. These subsidiaries were intended to own the American Centrifuge Plant (ACP) and equipment, provide operations and maintenance services, manufacture centrifuge machines and conduct ongoing centrifuge research and development. These subsidiaries are American Centrifuge Holdings, LLC (ACH), a direct subsidiary to Centrus, and ACO; American Centrifuge Technology, LLC (ACT); American Centrifuge Manufacturing, LLC (ACM); and American Centrifuge Enrichment, LLC (ACE), direct subsidiaries to ACH. ACO is the licensee and operating organization for the ACP. ACO will operate the HALEU Demonstration Program under the NRC ACP license.

Due to the current oversupply in the enrichment market, Centrus does not plan for near term deployment of a commercial scale uranium enrichment facility. As a result, Centrus has consolidated the ACP operations in Piketon, Ohio, and the technical, engineering and manufacturing capabilities in Oak Ridge, Tennessee, into ACO. Currently ACH, ACT, ACM and ACE are inactive companies.

ACO's principal officers are the same as Centrus' principal officers. The officers of ACO are citizens of the United States.

The Licensee holds the regulatory licenses and permits, including the NRC license, required to construct and operate the centrifuge facilities in Piketon, Ohio. The workers necessary to operate the centrifuge facilities in Piketon will be employed by the Licensee or its qualified contractors. Contracted resources are utilized in a number of these programmatic areas to provide day-to-day functional support. Inter-company arrangements (i.e., through reverse work authorizations) are in place to provide the necessary support.

The mailing address for the Licensee at the ACP is:

American Centrifuge Operating, LLC
American Centrifuge Plant
P. O. Box 628
Piketon, Ohio 45661-0628

1.2.1.1 Site Location

The ACP is located on DOE-owned land in rural Pike County, a sparsely populated area in south-central Ohio. Specifically, the ACP is located on the DOE reservation in the former GCEP facilities. The buildings/facilities and grounds are leased by Centrus from the DOE. The Licensee in turn subleases the buildings and grounds from Centrus. The DOE reservation has been studied and characterized extensively by both the DOE and Centrus.

The United States Enrichment Corporation, leases portions of the Portsmouth GDP reservation from the DOE. Pursuant to a 2006 amendment to that lease agreement, Centrus, formerly known as USEC Inc., subleased space for the Lead Cascade and the ACP from the United States Enrichment Corporation. Centrus, with approval of the DOE, assigned the sublease for the space for the ACP to the Licensee. The Licensee and its agents will conduct activities within the leased facilities and access and egress thereto, in accordance with this license application.

1.2.1.2 Other Reservation Activities

In addition to the Licensee's operations, the DOE operates a depleted uranium hexafluoride (DUF_6) Conversion Facility on the reservation adjacent to the ACP. The DOE is also engaged in activities related to the decontamination and decommissioning (D&D) of the GDP and environmental restoration activities in a number of locations on the reservation. DOE utilizes contractors and sub-contractors to perform this work. DOE self-regulates DOE activities conducted in non-leased areas in accordance with applicable DOE requirements.

Mid-America Conversion Services, LLC (MCS) currently manages the DUF_6 Conversion Facility at the DOE reservation. The DUF_6 Conversion Facility was designed and constructed to convert DOE's inventory of DUF_6 produced by the former Portsmouth GDP to a more stable uranium oxide form for reuse, storage, and/or transportation and disposition. The process also produces hydrogen fluoride (HF) as a conversion co-product. Excess HF is neutralized to calcium fluoride (CaF_2) (References 2 and 28). The DUF_6 area consists of cylinder storage yards, a process building, support buildings, a warehouse and an administration building.

Fluor-BWXT Portsmouth, LLC (FBP) is the DOE contractor for D&D of the GDP. FBP is responsible for the D&D of 415 facilities and structures that supported the uranium enrichment operations conducted at the site. During D&D, Fluor-BWXT prepares contaminated facilities for demolition by deactivating utilities and removing stored waste, materials, process equipment such as converters and compressors, and piping.

The plant also includes various support structures that provide feed and transfer operations and site services such as maintenance; steam generation; cleaning; process heat removal; electrical power distribution; and water supply storage and distribution.

Pixelle Specialty Solutions™, formerly Glatfelter Specialty Papers, operates a lumberyard on the north edge of the DOE reservation. This facility is utilized as a sorting and transfer area for commercial and paper grade lumber.

1.2.2 Financial Qualifications

Under the HALEU Contract (Reference 17), DOE agreed to reimburse the Company for 80 percent of its costs incurred in performing the contract. The Company's cost share is the corresponding 20 percent and any costs incurred above these amounts. Costs under the HALEU Contract include *program costs*, including direct labor and materials and associated indirect costs that are classified as *Cost of Sales*, and an allocation of corporate costs supporting the program that are classified as S*elling, General, and Administrative Expenses*. Services to be provided over the three-year contract include constructing and assembling centrifuges and related infrastructure in a cascade formation. When estimates of remaining program costs to be incurred for such an integrated construction-type contract exceed estimates of total revenue to be earned, a provision for the remaining loss on the contract is recorded to *Cost of Sales* in the period the loss is determined. Our corporate costs supporting the program are recognized as expense as incurred over the duration of the contract term. The accrued loss on the contract will be adjusted over the remaining contract term based on actual results and remaining program cost projections (Reference 22).

In support of this HALEU Demonstration Program, DOE amended the GCEP Lease Agreement, in which the parties agree that all work performed under the HALEU Demonstration Contract on leased premises shall be considered a permitted use; any alterations or changes to the premises pursuant to the Demonstration Contract with the DOE shall be a permitted change to the premises; and that any liabilities of the Corporation (Licensee) arising from or incident to the performance of work under the Demonstration Contract with the DOE shall be governed solely by such contract. Both the GCEP Lease and the Demonstration Contract afford indemnification pursuant to the Price Anderson Act.

The Company has long-term nuclear fuel sales and supply contracts in place that extend to 2030; these contracts will provide a stream of revenue for many years and provide a foundation for growth (Reference 22).

At the time of initial licensing and remains as the basis for the initial Materials License approval, the Licensee estimated the total cost to construct the initial 3.8 million SWU capacity for the ACP to be up to $3.1 billion (2008 dollars) (Reference 3) (see Appendix C of this license application), excluding capitalized interest, tails disposition, decommissioning, and any replacement equipment required during the life of the plant outside of normal spare equipment. The commercial ACP design is modular and can be constructed and installed incrementally over time. As the final commercial ACP phase, the Licensee plans to construct the plant and install centrifuges in increments until the ACP reaches a capacity of up to 3.8 million SWU production annually. As groups of centrifuges are installed, operations will be initiated and will result in enrichment production that will generate revenue. The Licensee may construct and install additional capacity thereafter as operations and market conditions permit subject to additional NRC licensing approval. Financing for each phase of incremental capacity may be raised using different financial instruments, and the ratio of equity to debt may vary over time for each increment.

Funding for various future phases of construction may come from a variety of sources including, but not limited to, funds from operations, capital raised by the Licensee, other American Centrifuge limited liability companies, lending and/or lease arrangements and that the mix of funding sources may vary depending upon the phase of the project. Prior to initiating each phase, the Licensee will make available for inspection on a confidential basis, its budget estimate for such phase and documentation of the source of funds available or committed to fund that increment.

In general, the Licensee's financial qualifications to construct and operate the HALEU 16-centrifuge cascade under the Demonstration Contract is demonstrated by the contract with DOE and the Selected Financial Data and detailed Consolidated Financial Statements within the latest information filed with the U.S. Securities Exchange Commission by its parent Centrus.

In order to meet the financial qualifications requirements for construction and operation of future expansion of the facility beyond the cascade funded under the HALEU Demonstration Contract, the Licensee proposes that the license be conditioned as follows:

- Construction of each additional incremental future expansion of the ACP shall not commence before funding for that increment is available or committed. Of this funding, the Licensee or affiliates must demonstrate before constructing such increment, arrangements that solely or cumulatively are sufficient to ensure funding for the particular increment's construction costs. The Licensee will make available for NRC inspection, documentation of both the budgeted costs for such phase and the source of funds available or committed to pay those costs.

- Operation of additional expansion of the ACP shall not commence until the Licensee or affiliates has in place, either: (1) long term contracts lasting five years or more that provide sufficient funding for the estimated cost of operating the facility for the five year period; (2) documentation of the availability of one or more alternative sources of funds that provide sufficient funding for the estimated cost of operating the facility for five years; or (3) some combination of (1) and (2).

Pursuant to Section 3107 of the USEC *Privatization Act*, the United States Enrichment Corporation leases the portions of the DOE reservation from DOE on which the ACP is located. The Licensee subleases those portions of the DOE reservations from the United States Enrichment Corporation. Under its lease with DOE and the sublease, and in accordance with Section 3107, the United States Enrichment Corporation and the Licensee are indemnified under Section 170d of the *Atomic Energy Act* for liability claims arising out of any occurrence within the United States, causing, within or outside the United States, bodily injury, sickness, disease, or death, or loss of or damage to property, or loss of use of property, arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source or special nuclear material arising out of activities under the lease. This indemnification is sufficient to meet the requirements of Section 193(d) of the *Atomic Energy Act* of 1954, as amended, and 10 CFR 140.13b, because the DOE indemnity provides greater financial protection than commercially available liability insurance. Therefore, the appropriate amount of separate liability insurance that should be required by the NRC is zero and an exemption from the requirements of 10 CFR 140.13b crediting DOE indemnity in lieu of nuclear liability insurance as discussed in this section is provided in Section 1.2.5 of this license application.

By letter dated May 14, 2007 (AET 07-0030) the Licensee provided status of its efforts to obtain nuclear liability insurance in accordance with NRC License Condition #14. The NRC agreed on July 16, 2007 that the Licensee had satisfied the requirements of this license condition and no further action is required concerning this license condition.

Information indicating how reasonable assurance will be provided that funds will be available to decommission the facility as required by 10 CFR 70.22(a)(9), 10 CFR 70.25, and 10 CFR 40.36 is described in Chapter 10.0 of this license application.

1.2.3 Type, Quantity, and Form of Licensed Material

The type, quantity, and form of NRC-regulated special nuclear, source, and by-product material are shown in Table 1.2-1 for the proposed commercial plant and Table 1.2-2 for the HALEU Demonstration Program (see Appendix D of this license application).

1.2.4 Authorized Uses

The commercial ACP operation enriches UF_6 up to 10 wt. percent ^{235}U. The specific authorized uses for each class of NRC-regulated material are shown in Table 1.2-3.

The HALEU Demonstration cascade enriches UF_6 up to a target enrichment of 19.75 wt. percent ^{235}U, but less than 20 wt. percent ^{235}U. Enrichment levels up to 25 wt. percent ^{235}U are authorized to permit for process fluctuations which can create small amounts of higher weight percent material. The specific authorized uses for each class of NRC-regulated material for the HALEU Demonstration Program are shown in Table 1.2-4. The Licensee proposes that the license be conditioned as follows:

ACP shall not enrich UF_6 in excess of 20 wt. percent ^{235}U other than in the course of cascade performance adjustments, thus providing the operational flexibility to generate material to satisfactorily fulfill customer orders up to 20 wt. percent ^{235}U. ACP shall not input parameters to extract product material for the assay above 20 wt. percent ^{235}U at any time.

Within the ACP Operations, the Licensee will provide a minimum 60-day notice to the NRC prior to initial customer product withdrawal of licensed material exceeding 5 wt. percent ^{235}U enrichment. This notice will identify the necessary equipment and operational changes to support customer product withdrawal, storage, processing, and shipment for these assays.

1.2.5 Special Exemptions or Special Authorizations

The following exemption to the applicable 10 CFR Part 20 requirements are identified in Section 4.8 of this license application:

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

- Containers located in Restricted Areas within the ACP are exempt from container labeling requirements of 10 CFR 20.1904, as it is deemed impractical to label each and every container. In such areas, one sign stating that every container may contain radioactive material will be posted. By procedure, when containers are to be removed from contaminated or potentially contaminated areas, a survey is performed to ensure that contamination is not spread around the reservation.

- In lieu of the requirements of 10 CFR 20.1601(a), each High Radiation Area with a radiation reading greater than 0.1 Roentgen Equivalent Man per hour (REM/hour) at 30-centimeters (cm) but less than 1 REM/hour at 30 cm is posted Caution, High Radiation Area and entrance into the area shall be controlled by an RWP. Physical and administrative controls to prevent inadvertent or unauthorized access to High and Very High Radiation Areas are maintained. The on-site radiological impacts from the proposed exemptions to the requirements of 10 CFR 20.1904 and 20.1601 would be minimal and are consistent with previously approved exemptions found in the GDP certification. Moreover, pursuant to the regulations in 10 CFR 20.2301, the requested exemption is authorized by law and would not result in undue hazard to life or property.

The following exemption from the applicable 10 CFR 70.50 reporting requirement is identified in Section 11.6.3 of this license application:

- The 10 CFR 70.50(c)(2) reporting criteria require that the ACP submit a written follow-up report within 30 days of the initial report required by 10 CFR 70.50 (a) or (b) or by 10 CFR 70.74 and Appendix A of Part 70. In lieu of the 30-day requirement described in 10 CFR 70.50(c)(2), NRC approval to submit the required written reports within 60 days of the initial notifications is hereby requested.

10 CFR 70.17 allows the Commission, upon application of any interested person or upon its own initiative, to grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The requested exemption is authorized by law because there is no statutory prohibition on extending the reporting period to 60 days.

Furthermore, granting this exemption request will not endanger life or property or the common defense and security, in that the exemption request does not relieve the ACP from other requirements contained in 10 CFR 70.50 (a) or (b) or by 10 CFR 70.74 and Appendix A of Part 70, such as 1-hour, 4-hour, and 24-hour reporting requirements for defined events.

The proposed exemption would result only in written reports being submitted within the time limit currently allowed under 10 CFR 50.73 for commercial nuclear power plants. It would be consistent with the exemption granted to the gaseous diffusion plants for reporting of events pursuant to 10 CFR 76.120(d)(2) (67 Federal Register 68699, November 12, 2002) and the exemption granted to the Lead Cascade during licensing.

This proposal allows for completion of required root cause analyses after event discovery and fewer supplemental reports, thereby reducing regulatory burden and confusion. Thus, it is clearly consistent with the public interest.

The Licensee notes that the requirements of 10 CFR 20.2201 and 20.2203 require written reports of certain events within 30 days after their occurrence. The Licensee is not requesting an exemption from these reporting requirements.

The following exemption from the requirements of 10 CFR 70.25(e) and 10 CFR 40.36(d) addressing the decommissioning funding requirements is identified in Section 10.1 of this license application:

- 10 CFR 70.25(e) and 10 CFR 40.36(d) require, in part, that "The decommissioning funding plan must also contain a certification by the licensee that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning…".

  In support of HALEU Demonstration Program, as noted in Section 10.1 of this license application, DOE amended the *Appendix 1 Lease Agreement between the U.S. Department of Energy and United States Enrichment Corporation for the Gas Centrifuge Enrichment Plant* (GCEP Lease Agreement). In the amended GCEP Lease Agreement, DOE assumes all liability for the decontamination and decommissioning of such facilities and equipment installed, and any work performed, under the Demonstration Contract with the Department including any materials or environmental hazards on the site. Therefore, exempting ACO from any financial assurance for any liability or lease turnover conditions shall be required from the Corporation (Licensee). Additionally, as stated within the amended GCEP Lease Agreement, the parties agree that should any liabilities of the Corporation (Licensee) arise from or incident to the performance of work under the Demonstration Contract with the DOE shall be governed solely by such contract and any financial protection afforded to the Corporation (Licensee) as a person indemnified under the Act.

The following exemption from the requirements of 10 CFR 70.25(e) and 10 CFR 40.36(d) addressing the decommissioning funding requirements is identified in Section 10.2.10.4 and the DFP of this license application:

- 10 CFR 70.25(e) and 10 CFR 40.36(d) require, in part, that "The decommissioning funding plan must also contain a certification by the licensee that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning…".

  In support of future expansion of the ACP, as noted in Section 10.2.10.4 of this license application, the financial assurance for a portion of the decommissioning costs, to include the disposition of centrifuges and UF_6 tails, which constitutes a major portion of the decommissioning liability, will be provided incrementally as centrifuges are built/installed and UF_6 tails generated. Full funding for decommissioning of the facilities will be provided in the initial executed financial assurance instrument.

  This exemption is justified for the following reasons: 1) It is authorized by law because there is no statutory prohibition on incremental funding of decommissioning costs. 2) The requested exemption will not endanger life or property or the common defense and security for the following reasons: the unique modular aspects of the American Centrifuge technology allow enrichment operations to begin well before the full capacity of the plant is reached. Thus, the decommissioning liability for centrifuges and UF_6 tails is incurred incrementally as more centrifuges are added to the process, until full capacity of the facility is reached; at which point the UF_6 tails are generated at a relatively constant rate throughout the life of the plant. As such, requiring full funding for decommissioning liability, to include centrifuges and UF_6 tails disposition, incurred over the lifetime of the plant, at the time of initial license issuance, produces an unnecessary financial burden on the licensee.

  Furthermore, incremental funding of decommissioning costs, to include centrifuges and UF_6 tails disposition, is justified based upon the Licensee's commitments to update the cost estimates and provide a revised funding instrument for decommissioning annually, to cover the upcoming period of operation, prior to operation at full capacity, and after full capacity has been reached to annually adjust the cost estimate for UF_6 tails disposition and to adjust all other decommissioning costs periodically, and no less frequently than every three years. In addition, the relative stability of the factors, which are utilized to generate the UF_6 tails volumes, allows actual inventory values to be provided for prior periods of operation and reliable estimates for the upcoming periods of operation. The NRC has previously accepted an incremental approach to decommissioning funding costs for the United States Enrichment Corporation's operation of the GDPs. 3) Finally, granting this exemption is in the public interest for the same reasons as stated above and will facilitate deployment of gas centrifuge enrichment technology by eliminating an unnecessary financial burden on the licensee.

The following exemption from the requirements of 10 CFR 70.24 addressing criticality monitoring is identified in Section 3.10.6 of the ISA Summary and discussed in Section 5.4.4 of this License Application. Exemption is required for criticality monitoring of the UF_6 cylinder storage yards.

- 10 CFR 70.24, *Criticality Accident Requirements*, requires that licensees authorized to possess special nuclear material in a quantity exceeding 700 g of contained ^{235}U shall maintain in each area in which such licensed special nuclear material is handled, used, or stored, a monitoring system capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of two meters from the reacting material within one minute.

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

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

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

The frequencies of criticality events in the cylinder yards have been decreased to the Highly Unlikely range (<10^{-5}/year) through the establishment of preventive controls established by the ISA in accordance 10 CFR 70.62. Considering the conservatism of the ISA methodology in developing the unmitigated frequency and actual historical data related to cylinder operations, the frequency values could be reduced further. This additional reduction considers the fact that during 50 years of GDP operations, only one cylinder breach has occurred due to mishandling or equipment failure. Since that occurrence, cylinder handling equipment has been redesigned and cylinder handling methods have been revised to minimize the potential for breaches to occur. Another fact not considered in the ISA is that holes with a dimension of less than one inch will self-seal such that moderating material cannot infiltrate the breach. A third factor not considered in the ISA is that enriched cylinder operations require constant use and monitoring of cylinders such that corrosion breaches in enriched cylinders are highly unlikely. Allowing for this additional reduction in frequency, the probability for a criticality event becomes incredible, therefore CAAS coverage is not necessary.

The increased vehicular and pedestrian traffic in support of CAAS maintenance and calibration requirements would cause a subsequent increased likelihood for impact events involving cylinders and there would be an increased safety risk for workers from radiation exposure due to the ongoing CAAS maintenance and calibration requirements. To meet the CAAS coverage requirements in ANSI 8.3 and the operating requirements for the ACP, enriched cylinder storage yards would require a minimum of 60 clusters. Clusters would need to be at a height of approximately 40 feet, which would require maintenance equipment and pedestrian traffic to perform testing and preventative maintenance tasks to ensure their reliability and operability. This equipment and traffic would increase the likelihood for fire and impact events in the cylinder storage yards such that workers would be at a higher risk for injury and exposure relative to the minimal mitigative value produced by the presence of CAAS.

The following exemption from the requirements of 10 CFR 140.13b crediting DOE indemnity in lieu of nuclear liability insurance as discussed in Section 1.2.2 of this license application.

- 10 CFR 140.13b requires, that "Each holder of a license issued under Parts 40 or 70 of this chapter for a uranium enrichment facility that involves the use of source material or special nuclear material is required to have and maintain liability insurance. The liability insurance must be the type and in the amounts the Commission considers appropriate to cover liability claims arising out of any occurrence within the United States that causes, within or outside the United States, bodily injury, sickness, disease, death, loss of or damage to property, or loss of use of property arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source material or special nuclear material. Proof of liability insurance must be filed with the Commission as required by § 140.15 before issuance of a license for a uranium enrichment facility under parts 40 and 70 of this chapter."

In support of this HALEU Demonstration Program, DOE amended the GCEP Lease Agreement, in which the parties agree that all work performed under the HALEU Demonstration Contract on leased premises shall be considered a permitted use; any alterations or changes to the premises pursuant to the Demonstration Contract with the DOE shall be a permitted change to the premises; and that any liabilities of the Corporation (Licensee) arising from or incident to the performance of work under the Demonstration Contract with the DOE shall be governed solely by such contract. Therefore, the Demonstration Contract exempts ACO from any financial assurance for any liability insurance during the three-year contract period.

In support of future expansion of the ACP, in accordance with Section 3107 of the USEC *Privatization Act*, the Lease with DOE for the DOE owned facilities that will be used for the ACP includes an indemnity agreement from DOE under Section 170d of the *Atomic Energy Act* (AEA) for liability claims.

The Commission may, pursuant to 10 CFR 140.8, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and are otherwise in the public interest. This exemption is authorized by law because there is no statutory prohibition on crediting the DOE indemnity agreement in lieu of nuclear liability insurance. The DOE indemnity agreement contained in the Lease pursuant to DOE's authority in Section 170d of the AEA is sufficient to meet the requirements of Section 193(d) of the *Atomic Energy Act* of 1954, as amended. Section 193(d) states that "the Commission shall require, as a condition of the issuance of a license ... for a uranium enrichment facility, that the licensee have and maintain liability insurance of such type and in such amounts as the Commission judges appropriate to cover liability claims ..."

The Lease requires that the Licensee obtain "financial protection to cover public liability, [as defined in the AEA] in such amount and of such type as is commercially available at commercially reasonable rates, terms and conditions" (Lease at Section 10.1(c)). To the extent required by the Lease, the Licensee will obtain such financial protection and will provide proof of such financial protection to the NRC prior to commencing operations.

The indemnity agreement contained in the Lease will "cover liability claims arising out of any occurrence within the United States that causes, within or outside the United States, bodily injury, sickness, disease, death, loss of or damage to property, or loss of use of property arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source material or special nuclear material." Section 193(d) affords the Commission the discretion to determine the type and amount of liability insurance that is required to cover liability claims. The

Commission has the discretion to conclude that no liability insurance is required in light of the DOE indemnity agreement. Therefore, the requested exemption is authorized by law.

Moreover, the requested exemption is in the public interest since it will facilitate deployment of the ACP, thereby maintaining domestic enrichment capacity using more efficient centrifuge technology. Requiring separate nuclear liability insurance would at best impose an unnecessary financial burden on the licensee and at worst preclude the construction of the ACP if commercial insurance ultimately is unavailable for facilities, such as the ACP, which are located on a DOE owned site. ANI, the only company providing commercial nuclear liability insurance in the U.S., has informed us that it has never insured a facility located on a DOE owned site. Furthermore, the separate liability insurance would not provide a commensurate benefit to the public since the DOE indemnity covers any public liability under Section 170 of the AEA up to the statutory limit of liability. The DOE indemnity agreement in the Lease adequately provides financial protection for the public for public liability as defined in the AEA. Therefore, the requested exemption is in the public interest.

The following exemption from NRC's Materials License Condition 15 related to financial funding as discussed in Section 1.2.2 of this license application.

- In order to meet the financial qualifications requirements for construction and operation of the facility, the Licensee proposes that the license be conditioned as follows:

  > Construction of each additional incremental future expansion of the ACP shall not commence before funding for that increment is available or committed. Of this funding, the Licensee or affiliates must demonstrate before constructing such increment, arrangements that solely or cumulatively are sufficient to ensure funding for the particular increment's construction costs. The Licensee will make available for NRC inspection, documentation of both the budgeted costs for such phase and the source of funds available or committed to pay those costs.
  >
  > Operation of additional expansion of the ACP shall not commence until the Licensee or affiliates has in place, either: (1) long term contracts lasting five years or more that provide sufficient funding for the estimated cost of operating the facility for the five year period; (2) documentation of the availability of one or more alternative sources of funds that provide sufficient funding for the estimated cost of operating the facility for five years; or (3) some combination of (1) and (2).

In general, the Licensee's financial qualifications to construct and operate the HALEU 16-centrifuge cascade under the Demonstrations' Contract is demonstrated by the contract with DOE and the Selected Financial Data and detailed Consolidated

Financial Statements within the latest information filed with the U.S. Securities Exchange Commission by its parent Centrus.

Under the HALEU Contract, DOE agreed to reimburse the Company for up to 80 percent of its costs incurred in performing the contract. The Company's cost share is the corresponding 20 percent and any costs incurred above these amounts. Costs under the HALEU Contract include *program costs*, including direct labor and materials and associated indirect costs that are classified as *Cost of Sales*, and an allocation of corporate costs supporting the program that are classified as *Selling, General, and Administrative Expenses*. Services to be provided over the three-year contract include constructing and assembling centrifuges and related infrastructure in a cascade formation and production of up to 600 kgU HALEU. When estimates of remaining program costs to be incurred for such an integrated construction-type contract exceed estimates of total revenue to be earned, a provision for the remaining loss on the contract is recorded to *Cost of Sales* in the period the loss is determined. Our corporate costs supporting the program are recognized as expense as incurred over the duration of the contract term. The accrued loss on the contract will be adjusted over the remaining contract term based on actual results and remaining program cost projections. The Licensee requests an exemption to this condition during the three-year HALEU Contract period.

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.

The following exemption from the requirements in 10 CFR 95.57(c) is identified in Section 1.17.c) of the *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant*:

- NRC regulations in 10 CFR 95.57(c) require that all classification actions (documents classified, declassified, or downgraded) to be submitted to the NRC Division of Security Operations. These may be submitted either on an "as completed" basis or monthly. The information may be submitted either electronically by an on-line system or by paper copy using NRC Form 790. Historically, the Licensee has utilized NRC Form 790 for each classification action, has compiled them monthly, and submitted them to the NRC. The Licensee must also submit a quarterly classification summary document to the DOE for all derivative classification decisions made during the previous quarter. This dual reporting is burdensome to the Derivative Classifiers and the Centrifuge Classification Officer and creates a situation where the classification actions may be double counted. Accordingly, in lieu of filing its classification actions with NRC, the Licensee will continue to submit the quarterly classification summary documents to DOE and will make them available for NRC inspection at the facility.

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 ACP, a plan describing the plant's proposed security procedures and controls, as set forth in 10 CFR Part 95, for the protection of classified matter. The Licensee satisfies the 10 CFR 70.22(m) requirements by submittal of the *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant*. The Security Plan was submitted for NRC review along with this license application. In accordance with 10 CFR Part 95.15(b), the Licensee will submit, at least 60 days prior to operation of the ACP, a request for a revision to the Facility Clearance from non-possessing facility to a possessing facility.

The Licensee shall provide the Commission with at least 120 days advance notice of its plan to introduce classified matter in the American Centrifuge Plant and the updated Security Plan for review and approval, consistent with 10 CFR Part 95 *Format and Content Guide*.

1.2.7 Security of Special Nuclear Material of Low Strategic Significance and Moderate Strategic Significance

Pursuant to 10 CFR 70.22(k) the Licensee is submitting, as part of its application for a license for the ACP, a plan describing the measures used to protect Special Nuclear Material of Strategic Significance that the Licensee uses, possesses, or has access to at the plant. The Licensee satisfies the 10 CFR 70.22(k) requirement by submittal of the *Security Plan for the Physical Protection of Special Nuclear Material at the American Centrifuge Plant*. The Security Plan is being submitted for NRC review along with this license application.

The specific design of the intrusion detection and alarm system is not yet complete. Upon completion of the design, the Licensee shall provide the Commission with at least 120 days advance notice of its plan to introduce special nuclear material in the American Centrifuge Plant, the final design for the intrusion detection and alarm system, and the Security Plan for review and approval, consistent with 10 CFR Part 95 *Format and Content Guide*.

Table 1.2-3 Commercial ACP Authorized uses of NRC-regulated materials	
Material Class	**Authorized Use**
A. Source Material, Element 92[a, b]	1. Enrichment of uranium up to 10 percent enrichment by weight ^{235}U
	2. Receipt, storage, inspection, acceptance, and sampling of cylinders containing uranium
	3. Filling and storage of cylinders of normal uranium and uranium depleted in ^{235}U
	4. Cleaning and inspection of cylinders used for the storage and transport of process product and tails containing source or Special Nuclear Material
	5. Storage of process wastes containing uranium, transuranic elements, and other contaminants and decay products
	6. Process, characterize, package, ship, or store low-level radioactive and mixed wastes
	7. Radiation protection, process control and environmental sample collection, analysis, instrument calibration, and operation checks
	8. Maintenance, repair, and replacement of process equipment
	9. Laboratory analysis and testing
	10. Heating cylinders and feeding contents into the enrichment process
	11. Transfer between cylinders
B. Source Material, Element 90	1. Calibration and use of portable radiation protection and fixed laboratory equipment
	2. Laboratory analysis and testing
	3. Process, characterize, package, ship, or store low-level radioactive and mixed wastes
C. Special Nuclear Material [a,b]	1. Filling, assay, storage, and shipment of cylinders and other Nuclear Criticality Safety approved containers containing uranium enriched up to 10 percent by weight ^{235}U
	2. Nondestructive testing and analyses of product and process streams

Table 1.2-3 Commercial ACP Authorized uses of NRC-regulated materials

Material Class	Authorized Use
	3. Receipt, storage, inspection, and acceptance sampling of cylinders containing uranium enriched up to 10 percent by weight ^{235}U
	4. Cleaning and inspection of cylinders used for the storage and transport of process feed, product, and tails containing source or Special Nuclear Material
	5. Storage of process wastes containing uranium, transuranic elements, and other contaminants and decay products
	6. Process, characterize, package, ship, or store low-level radioactive and mixed wastes
	7. Radiation protection, process control and environmental sample collection, analysis, instrument calibration, and operation checks
	8. Maintenance, repair, and replacement of process equipment
	9. Laboratory analysis and testing
	10. Heating cylinders and feeding contents into the enrichment process
	11. Transfer between cylinders
	12. Material remaining in cylinders and facilities as a result of previous operations
D. By-product Material, Elements 3-89, 91	1. Radiation protection, process control, and environmental sample collection, analysis, instrument calibration, and operation checks
	2. Laboratory analysis and testing
	3. Nondestructive testing of product and product streams
	4. Storage of process wastes containing uranium, transuranics, process contaminants, and decay products
	5. Material remaining in equipment and facilities as a result of feeding reprocessed uranium
	6. Process, characterize, package, ship, or store low-level radioactive and mixed wastes [C]

Table 1.2-3 Commercial ACP Authorized uses of NRC-regulated materials

Material Class	**Authorized Use**
Elements 93, 95 to 100	1. Calibration and use of portable radiation protection and fixed laboratory equipment
	2. Laboratory analysis and testing
	3. Nondestructive testing of product and product streams
	4. Storage of process wastes containing uranium, transuranics, process contaminants, and decay products
	5. Material remaining in cylinders and facilities as a result of feeding reprocessed uranium
	6. Process, characterize, package, ship, or store low-level radioactive and mixed wastes[c]
$^{43}{}_{99}Tc$	1. Material remaining in cylinders and facilities as a result of feeding reprocessed uranium
	2. Storage of process wastes as a result of feeding reprocessed uranium

[a] Uranium to be fed to the enrichment plant will meet the requirements of ASTM Standard C996, "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5% ^{235}U or ASTM standard C787, "Standard Specification for Uranium Hexafluoride for Enrichment" for reprocessed UF_6. Other uranium that does not meet the requirements of ASTM C996 or C787 for reprocessed UF_6 may be accepted for storage and subsequent disposition but will not be introduced to the enrichment process, with the exception of small amounts (e.g., 50 pounds UF_6) associated with sampling, subsampling, and analyses required to establish receiver's values.

[b] Includes the feed and processing of Paducah Product and any "stockpile" UF_6 transferred from DOE to the Licensee for enrichment.

[c] Includes the potential return of material (waste) generated at the ACP, sent off-site, and subsequently returned.

Table 1.2-4 HALEU Demonstration Program Authorized uses of NRC-regulated materials

Material Class	Authorized Use
A. Uranium (non-fissile) and daughter products 92[a, b]	1. Activities involving uranium enriched to less than 1.0 wt.% ^{235}U
	2. Receipt, storage, inspection, acceptance, and sampling of cylinders containing uranium
	3. Filling and storage of cylinders of normal uranium, depleted, and uranium enriched to less than 1.0 wt.% ^{235}U
	4. Storage of process wastes containing uranium, transuranic elements, and other contaminants and decay products
	5. Process, characterize, package, ship, or store low-level radioactive and mixed wastes
	6. Radiation protection, process control, environmental sample collection, instrument calibration, and operation checks
	7. Maintenance, repair, and replacement of process equipment
B. Source Material, Isotopes and Other Contamination Element 90	1. Calibration and use of portable radiation protection and fixed laboratory equipment
	2. Activities required to obtain samples for analysis whether on-site or off-site, and the potential subsequent return of this material for disposition (waste, utilization).
	3. Process, characterize, package, or store low-level radioactive and mixed wastes
C. Special Nuclear Material [a,b]	1. Feeding cylinders enriched to up to 5 percent by weight ^{235}U, and filling cylinders containing enriched material less than 20 percent by weight ^{235}U.
	2. The HALEU cascade is operated at less than 20 weight percent ^{235}U. Enrichment levels up to 25 weight percent ^{235}U are authorized to permit for process fluctuations which can create small amounts of higher weight percent material.
	3. Receipt, storage, inspection, acceptance, and sampling of cylinders and other Nuclear Criticality Safety approved containers containing uranium enriched up to 20 percent by weight ^{235}U
	4. Nondestructive testing and analyses of product and process streams

Table 1.2-4 HALEU Demonstration Program Authorized uses of NRC-regulated materials

Material Class	**Authorized Use**
	5. Storage of process wastes containing uranium, transuranic elements, and other contaminants and decay products 6. Process, characterize, package, ship, or store low-level radioactive and mixed wastes 7. Radiation protection, process control, environmental sample collection, instrument calibration, and operation checks 8. Maintenance, repair, and replacement of process equipment 9. Activities required to obtain samples for analysis whether on-site or off-site, and the potential subsequent return of this material for disposition (waste, utilization). 10. Feeding contents into the enrichment process 11. Filling and storage of cylinders as enriched up to, but less than, 20 percent by weight ^{235}U.
D. By-product Material, Elements 3-89, 91	1. Radiation protection, process control, environmental sample collection, instrument calibration, and operation checks 2. Activities required to obtain samples for analysis whether on-site or off-site, and the potential subsequent return of this material for disposition (waste, utilization). 3. Nondestructive testing of product and product streams 4. Storage of process wastes containing uranium, transuranics, process contaminants, and decay products 5. Material remaining in equipment and facilities as a result of feeding reprocessed uranium 6. Process, characterize, package, or store low-level radioactive and mixed wastes[c]

Table 1.2-4 HALEU Demonstration Program Authorized uses of NRC-regulated materials

Material Class	Authorized Use
Elements 93, 95, to 100	1. Calibration and use of portable radiation protection and fixed laboratory equipment 2. Activities required to obtain samples for analysis whether on-site or off-site, and the potential subsequent return of this material for disposition (waste, utilization). 3. Nondestructive testing of product and product streams 4. Storage of process wastes containing uranium, transuranics, process contaminants, and decay products 5. Process, characterize, package, or store low-level radioactive and mixed wastes[c]
$^{43}_{99}Tc$	1. Material remaining in cylinders and facilities as a result of feeding operations 2. Storage of process wastes as a result of feeding operations.

[a] Uranium to be fed to the enrichment plant will meet the requirements of ASTM Standard C996, "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5% ^{235}U or ASTM standard C787, "Standard Specification for Uranium Hexafluoride for Enrichment."

[b] Includes the feed and processing of Paducah Product.

[c] Includes the potential return of material (waste) generated at the HALEU Demonstration Program, sent off-site, and subsequently returned.

1.3 Site Description

This section presents information on the ACP's location, geography, demographics, meteorology, surface hydrology, subsurface hydrology, geology, and seismology.

The ACP is located on DOE-owned land in rural Pike County, a sparsely populated area in south-central Ohio. Specifically, the ACP is located on the DOE reservation in the former GCEP facilities (Figure 1.1-1, located in Appendix B). The buildings and grounds are leased by Centrus Energy Corp. from the DOE. The Licensee in turn sub-leases the buildings and grounds from Centrus. The reservation has been studied and characterized extensively by both DOE and Centrus.

1.3.1 Geography

The DOE reservation is approximately 3,700 acres located on the east side of the Scioto River, near Piketon, Ohio, and approximately equidistant between Portsmouth and Chillicothe, Ohio. A topographic map of the reservation is provided in Figure 1.3-1.

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

The primary roadways near the DOE reservation are U.S. Route 23 and State Route 335, which traverse a roughly north-south course, and State Route 124 (same as State Route 32), which traverses an east-west course just north of the reservation.

The Pike County Airport is located approximately 11 miles north-northeast of the DOE reservation. No commercial flights or cargo shipping occurs there. The 4,900-ft runway supports single and twin-engine planes and small jets. The Greater Portsmouth Regional Airport, located approximately 15 miles southeast of the DOE reservation, provides only light plane service (Class 1 airport). The Chillicothe-Ross County Airport is located approximately 35 miles north-northeast of the DOE reservation. The nearest commercial airports are John Glenn Columbus International Airport in Columbus, Ohio, approximately 75 miles north, Rickenbacker Airport near Columbus, Ohio approximately 60 miles away, the Tri-State Airport in Huntington, West Virginia approximately 65 miles southeast, and the Cincinnati/Northern Kentucky International Airport, approximately 100 miles west.

1.3.2 Demographics

The DOE 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 reservation 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 Population

The DOE reservation worker population was 2,336 as of January 2020, but these workers are unequally distributed and reside in the surrounding counties. The nearest residential center and the closest town to the reservation is Piketon, located in Pike County about four miles north of the reservation on U.S. Route 23 with a population of 2,181 in 2010. The largest town in Pike County is Waverly, about eight miles north of the reservation, with a population of 4,408 in 2010. Chillicothe, in Ross County about 27 miles north, is the largest population center in the Region of Influence with a population of 21,698 in 2010. Other population centers include Portsmouth, about 27 miles south in Scioto County, and Jackson, about 26 miles east in Jackson County, with populations of 20,340 and 6,242 in 2010, respectively. Table 1.3-1 presents historic and projected population in the Region of Influence and the state. (References 4 and 34). The total population within the five-mile radius of the reservation was 5,805(Figure 1.3-2) in 2010. (Population information was obtained from census data - Reference 35).

1.3.2.2 Significant Transient and Special Populations

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

1.3.2.2.1 Schools

There are a number of educational institutions inside a five-mile radius of the DOE reservation. All of the Scioto Valley Local School District's (SVLSD) schools are within the five-mile radius. As of January 2020, these schools are the Piketon High School and Junior High School, located in the same building with 492 students and 27 teachers, Zahn's Corner Middle School with 303 students and 18 teachers (relocated to Piketon High School and Jasper Elementary for the 2019-2020 school year); and Jasper Elementary School with 385 students and 18 teachers (Reference 36). In addition to the SVLSD there is the Pike County Career Technology Center with 400 vocational high school students and adult education students, and 70 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 267students and 63 staff. In addition, there is a private pre and elementary school, Miracle City Academy, with 32 students and 5 staff (Reference 37). The locations and student-occupancies of these facilities are shown in Figure 1.3-3 (Reference 5).

1.3.2.2.2 Hospitals and Nursing Homes

Adena Pike Medical Center 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, approximately 147 total staff, and operates at full capacity. Adena Health Center operates an urgent care facility located in Waverly approximately 1 mile north of the hospital. The Southern Ohio Medical Center Family Health Center also operates an urgent care center in Waverly. The Valley View Health Center is located next to the Adena Pike Medical Center. The Adena Family Medicine – Piketon and, another Valley View Health Center are both located in Piketon.

There are two licensed nursing homes in the Piketon area, Piketon Nursing Center, and Pavilion at Piketon. As of January 2020, the Piketon Nursing Center had 46 patients and 46 staff, and the Pavilion at Piketon had 193 patients and 220 staff. Additionally, a home for people with intellectual and developmental disabilities in Wakefield, Scioto Trails Group Home, with 32 beds and 100 staff. Figure 1.3-3 depicts these medical and nursing facilities and shows the number of beds per facility (Reference 5).

1.3.2.2.3 Recreational Areas and Recreational Events

No significant recreational areas are located on the DOE reservation; recreational activities for employees are held off-site.

Off-site recreational areas include the Brush Creek State Forest, a 0.5 square mile portion of which is within five miles southwest of the 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 38).

Usage of Lake White State Park (Figure 1.3-3), located approximately six miles north of the reservation, is occasionally heavy and concentrated on the 92 acres of land closest to the lake. Most of the land surrounding the lake is privately owned. The 333-acre Lake White offers recreation, such as, boating, fishing, water skiing, and swimming. (Reference 10).

Rock Water Campground is a private, secured campground with 68 campsites within five miles west of the site. The site is approximately 20 acres that includes a 12 acre lake for swimming and fishing (Reference 39).

1.3.2.3 Uses of Nearby Lands and Waters

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

The only significant industry in the vicinity is located in an industrial park south of Waverly. The industries include a a farm supply store and distribution center, a plastic recycling and processing center, and an automotive parts manufacturer. These industries do not present any potential hazards to ACP operations.

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

No known public or private water is withdrawn from the Scioto River downstream of the ACP (Reference 40).

1.3.3 Meteorology

This section provides a meteorological description of the DOE reservation and its surrounding area. The purpose is to provide meteorological information necessary to understand the regional weather phenomena of concern for the ACP operations and to understand the basis for the dispersion analyses performed (Reference 41).

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. July is the hottest month, with an average monthly temperature of 75.0°F, and January is the coldest month with an average temperature of 29.9°F. The highest and lowest daily temperatures from 1951 to 2019 were 103°F and –31°F on July 14, 1954, and January 19, 1994, respectively (References 8, 13, 32 and 33).

Moisture in the area is predominantly supplied by air moving northward from the Gulf of Mexico. 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 2019 was 40 inches (in.). The greatest daily rainfall during this period was 4.9 in., occurring on March 2, 1997 (Reference 13).

Occasionally, heavy amounts of rain associated with thunderstorms or low-pressure systems falls in a short period of time. The Midwestern Climate Center, Climate Analysis Center, the National Weather Service, the National Oceanic and Atmospheric Administration, and the Illinois State Water Survey Division of the Illinois Department of Energy and Natural Resources have 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 reservation are summarized in Table 1.3-2 (Reference 13). A local drainage analysis for extreme storms at the site has also been performed (Reference 42).

Snowfall occurrence varies from year to year, but is common from November through March. The average annual snowfall for the area is about 21.1 in., based on 1951-2019 data. During that time period, the maximum monthly snowfall was 25.4 in., occurring in January 1978 (References 13 and 32). The design basis snowfall for building construction is the historical maximum snowfall, which equates to approximately 20 pounds per square foot (psf) and complies with standard ASCE-7-2002, *Minimum Design Loads for Buildings and Other Structures* (Reference 73).

1.3.3.2 On-Site Meteorological Measurements Program

A 60-m meteorological tower is used on the DOE reservation. The tower is equipped with instrument packages at the 10-, 30-, and 60-m (33-, 98-, and 197-ft) levels to measure the air temperature, wind speed, and wind direction. Other instrumentation measures the solar radiation, barometric pressure, precipitation, and soil temperatures.

1.3.3.3 Local Meteorology

Since January 1995, a 60-m (197-ft) tower has been in use. It is equipped with instrument packages at the 10-, 30-, and 60-m (33-, 98-, and 197-ft) levels. In addition, ground-level instrumentation measures solar radiation, barometric pressure, precipitation, and soil temperatures at 1 and 2-ft depths.

Hourly temperatures at the 10- and 30-m (33- and 98-ft) levels above the ground have been recorded at the site meteorological tower from since at least 1995. Data from the 1995 to 2002 period show that at the 10-m (33-ft), 69,734 of the possible 70,080 data points are available. At the 10-m level the average annual hourly temperature was 50.6°F, the minimum average hourly temperature was –1.4°F, and the maximum average hourly temperature was 94.1°F (Reference 6).

Of the 70,080 possible hourly wind speed and wind direction data for 1995 through 2002, approximately 70,000 are available points. Wind roses for the 10-, 30-, and 60-m (33-, 98-, and 197-ft) levels at the reservation constructed from the 1998 through 2002 data are compared in Figures 1.3-4, 1.3-5, and 1.3-6, respectively (Reference 6). The prevailing wind directions are from the south-southwest to southwest at the 10-m (33-ft) level.

Additional data from calendar year 2016 was also obtained. The average wind speeds were 3.6, 5.0, and 6.5 mph at the 10-, 30- and 60-meter levels, respectively. At the 10-meter level, the minimum average hourly temperature was 4.0 °F, and the maximum average hourly temperature was 96.4 °F.

Tornadoes do occur in Southern Ohio; however, specific analyses of the frequency of tornadoes in the region show that they are rare. On the average, from 1950 to 2010, 19 tornadoes per year were reported in Ohio, but the total varies widely from year to year (e.g., 63 in 1992 and 4 in 2005). Pike County has experienced eleven tornadoes since 1950. When considering the surrounding counties (Adams, Jackson, Highland, Ross, and Scioto), the total number of tornadoes experienced is 54 since 1950. Of those tornadoes, 12 were rated F2 or greater on the Fujita Tornado Scale (Reference 43). The reservation had an average of three days per year between 1990 and 2019 with severe storms with winds exceeding 58 mph (Reference 43). Because the reservation 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. The wind loading design basis criteria is provided in Section 3.4. The wind design loads used in the original design of the existing structures was prescribed in K-DA-603 (Reference 74). For new construction complying with standard ASCE-7-2002, *Minimum Design Loads for Buildings and Other Structures*, 7 psf/sec is the minimum design wind load.

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. The DOE reservation had an average of three days per year between 1990 and 2019 with severe storms with winds exceeding 58 mph, defined as severe thunderstorm winds. (Reference 43). The reservation 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.

1.3.4 Surface Hydrology

This section describes the surface hydrology on and around the DOE reservation.

1.3.4.1 Hydrologic Description

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

1.3.4.1.1 Scioto River Basin

The DOE 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 Scioto 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 44).

Upstream retarding basins are located on tributaries throughout the Scioto River basin. The upstream retarding basin nearest the reservation forms Lake White along Pee Pee Creek, about six miles north of the reservation (Figure1.3-7). The spillway of the reservoir is located at an elevation of 567 ft above mean sea level (amsl), while the roadway along the top of the dam is at an elevation of 577 ft amsl (Reference 45). 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 reservation 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 to2018 range from 177,000 cubic feet per second (cfs) on January 23, 1937, to 244 cfs on October 23, 1930. The annual mean flow has ranged from 1,364 cfs in 1954 to 8,178 cfs in 1996. The 1937 flood had a peak water elevation of 593.7 ft amsl. The consecutive seven-day minimum discharge of record is 255 cfs, which occurred during October 19-25, 1930 (References 46 and 47).

Water in the vicinity of the reservation is available from Lake White, the Scioto River, and groundwater supplies (Reference 48). 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 three water suppliers 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 49).

Water used at the reservation normally comes from groundwater. Currently, 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 DOE Reservation Area

The DOE reservation is located about 2 miles east of the confluence of the Scioto River and Big Beaver Creek near RM 27.5 (Figure 1.3-7). The reservation 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 amsl, which is about 130 ft above the normal stage of the Scioto River. Both groundwater and surface water at the reservation are drained from the plant 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 reservation. Little Beaver Creek, the largest stream on the property, flows northwesterly through the northern portion of the main plant area (Figure 1.3-7). It drains the northern and northeastern parts of the main plant 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 1.3-7). Upstream from the plant, Little Beaver Creek has intermittent flow throughout the year.

In the southeast portion of the reservation, the southerly flowing Big Run Creek (Figure 1.3-7) is situated in a relatively broad, gently sloping valley where significant deposits of recent alluvium have been laid down by the stream (Reference 50). This intermittent stream receives overflow from the X-230K South Holding Pond, which collects discharge of storm sewers on the south end of the plant. Big Run Creek empties into the Scioto River about five miles downstream from the mouth of Big Beaver Creek (Figure 1.3-7).

Two streams drain the western portion of the reservation (Figure 1.3-7). The stream in the plant's southwest portion flows southerly and westerly in a narrow, steep-walled valley with little recent alluvium. It drains the southwest corner of the ACP via the southwest holding pond. The stream near the west central portion of the reservation flows northwesterly and receives runoff from the central and western part of the reservation via the west drainage ditch. Both streams flow directly to the Scioto River and carry predominately storm water runoff, with lesser contributions from such sources as groundwater infiltration, steam condensate, and firewater (Reference 50).

Little Beaver Creek receives 39 percent of the total 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 51).

1.3.4.1.3 Site and Facilities

The DOE reservation nominal elevation is 670 ft amsl, which is about 130 ft above the normal stage of the Scioto River. The top-of-slab floor elevations for the ACP facilities are at approximately 671 ft amsl. Storm water that falls at the reservation 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 sewers.

The Perimeter Road, as shown in Figure 1.3-8, serves as a hydrologic boundary that prevents storm water runoff from backing up into the ACP. 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 ACP 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 1.3-8, to the topographic elevations, indicated in Figure 1.3-9, emphasizes the overall function of the reservation surface water drainage system that has been described here (Reference 42).

Water used at the reservation is supplied by wells sunk into the Scioto River alluvium. The raw water is pumped from wells at three locations along the Scioto River along with a backup system that can draw directly from the Scioto River when the wells are unable to produce sufficient water to meet the reservation demand. The well fields and pump house are located where flooding is anticipated, so the equipment is designed and installed to operate without adverse effect (Reference 48). The equipment in the pump house is located above the 571 ft amsl level and the well pumps can operate under water.

1.3.4.2 Flood History

The average annual discharge at the Higby station for the period of record (1930-2018) is 4,721 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 amsl. The historical flood stage of the Scioto River next to the DOE reservation was estimated to be 556.7 ft amsl 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 reservation nominal grade elevation in Table 1.3-3 (References 38, 46, 52, and 53).

Since the reservation has a nominal elevation of about 670 ft amsl (Figure 1.3-9) and about 113 ft above the historical flood level for the Scioto River in the area, the reservation 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 the whole basin is 6,517 square miles (Reference 52). The drainage area of the Scioto River above the DOE reservation (RM 27.5) is between those two values. A conservative estimate for the PMF discharge of the Scioto River at either Higby or the reservation is approximately 1,000,000 cfs. This value is used as the PMF discharge of the Scioto River at the reservation, which including the wind/wave activity contribution, would correspond to a flood level of 571 ft amsl, well below the nominal 670 ft amsl elevation of the reservation.

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 reservation.

Conservative estimates indicate that the failure of upstream dams would not threaten the safety of the reservation because of the high nominal plant grade elevation (Reference 54). In addition, the limited storage capacities of the reservoirs, the large stream distances of these dams from the reservation, and friction and form losses would make the actual wave heights even smaller than the estimated values. Discharges were considered for 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 operation of the reservation 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 Midwestern Climate Center, National Weather Service, National Oceanic and Atmospheric Administration, and Illinois State Water Survey Division of the Illinois Department of Energy and Natural Resources have 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 for the midwestern states, including Ohio (Reference 9). The results for the geographic locale including the DOE reservation 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 reservation is first 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. The resultant graph is shown in Figure 1.3-10. 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 reservation 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 1.3-10. Local flooding for different streams is caused by 10,000-year storms with differing duration values because each watershed drains a basin of a different size (Reference 42). The relatively large differences between nominal plant grade elevation and the calculated flood stage elevations for the five streams clearly indicate that the ACP 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 42) 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 reservation 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 55).

The effect of a clogged storm sewer system on the ponding depth has been considered (Reference 42). 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 reservation is not expected to impact the ACP safe operations.

Results for Ponds and Lagoons

To assess whether failures of the local dams could conceivably jeopardize the safety of ACP operations, 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, Storm Sewer L, and X-230L North Holding Pond (Reference42). The surface elevations of the reservation facilities are well below the 670-ft amsl minimum grade elevation of the ACP facilities.

Results for Ditches and Culverts

The reservation 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 reservation have been analyzed to predict their response during a 10,000-year storm (Reference 42). 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 the related culverts are below grade elevation at the ACP and would not cause local flooding at these buildings during a 10,000-year storm.

Effects of Ice and Snow

The reservation has a generally moderate climate. Winters in the area are moderately cold. On the average, there are 123 days per year below 32°F, but only approximately four days per year at or below 0°F. The average annual snowfall is 22 in. To estimate the extreme snowfall at the reservation, 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 reservation, and if an average density of 0.1 for freshly fallen snow is assumed (References 8 and 56), 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 as guidance to estimate the PMF discharge (Reference 14). 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 Scioto River is 6,517 square miles. The drainage area of the Scioto River above the DOE reservation (RM 27.5) is estimated from these values to be 6,000 square miles. PMF discharge of the Scioto River at the reservation as taken from the log-log plot is approximately 1,000,000 cfs. This value is adopted as the PMF discharge near the reservation (Reference 55).

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 reservation in accordance with Alternatives I and II in Appendix A of NRC Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants* (Reference 14). The fetch length near the DOE reservation 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 reservation due to this wind wave activity was estimated to be 1.8 ft (Reference 57). The PMF plus this coincident wind wave activity would have a flood stage of 571 ft amsl.

Comparison of Flood Levels with DOE Reservation Elevations

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

The reservation water supply facilities are located near the Scioto River. The X-608 Raw Water Pump House equipment is located just above the 571 ft amsl flood stage. The X-605G, X-608A, X-608B, and X-6609 Raw Water Wells are located below the 571 ft amsl flood stage, but are designed to operate during flood conditions (Reference 48).

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 amsl. However, even when a conservative wave height of 41.3 ft is used, this cascade of dam failures clearly would not threaten the DOE reservation because the nominal plant grade elevation is 670 ft amsl, 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 58). 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 59). 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 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 DOE reservation because the plant uses groundwater taken near the river. Additionally, ice formation would not pose a threat of flooding to the reservation, given the high elevation of the plant relative to the river.

1.3.4.6 Low Water Considerations

Water used at the DOE reservation can be supplied from wells in the Scioto River alluvium and pumped via existing waterlines to the X-611 Water Treatment Plant. The X-608 Pump House 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 48).

At the Higby gauging station, which is approximately 13 miles north of the reservation, the minimum river flow measured from 1930 to 2019 was 244 cfs on October 23, 1930. The consecutive seven-day minimum discharge record of 255 cfs occurred during October 19-25, 1930. (Reference 46). The volumetric river flow is much greater than the reservation's water use.

1.3.4.7 Dilution of Effluents

The average discharge of the Scioto River near the DOE reservation is 4,721 cfs. Potentially, this discharge rate has a large capacity for reducing the concentration of received contaminants. For example, the uranium discharged from the reservation from the GDP through the local drainage system to the Scioto River was estimated to be 45 kg during 1990 (Reference 60). In 1990, the bulk of the uranium (76 percent) was discharged through Outfall 001 to Little Beaver Creek (Reference60). Assuming a full dilution, this would result in an average uranium concentration of 1.1 x 10^{-5} milligrams per liter in the Scioto River well below the maximum concentration.

In support of ACP operations, the GDP NPDES permits have been modified to transfer ownership of certain discharge points. The Licensee now has two outfalls that discharge directly to surface water and one outfall that discharges to the FBP X-6619 Sewage Treatment Plant before leaving site through FBP Outfall 003 to the Scioto River. The Tower Water Cooling system discharges its blowdown to GDP Recirculating Cooling Water system under a service agreement, which in turn discharges its blowdown directly to the Scioto River via an underground pipeline (NPDES Outfall 004). FBP has eight outfalls and nine internal outfalls. MCS has one outfall and one internal outfall. In 2017, the overall Licensee's NPDES compliance rate was 100 percent and the overall FBP's NPDES compliance rate was 99 percent, with further details being provided in FBP-ER-RCRA-WD-RPT-0288 (Reference 70). Further description of Surface Water contaminants can be found in Section 3.4.2 of the Environmental Report.

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 DOE reservation.

1.3.5.1.1 Regional and Area Characteristics

In the region surrounding the DOE 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 DOE reservation is composed of unconsolidated Pleistocene clastic sediments of glacial and alluvial origin in river valleys and of underlying Paleozoic bedrock units. Figures 1.3-11 and 1.3-12 show the general configuration of these valleys and bedrock units near the reservation.

The unconsolidated sediments aquifer consists of two distinct aquifers in the immediate vicinity of the reservation: the Scioto River glacial outwash aquifer and "other" alluvial aquifers, 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 for 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 reservation. The bedrock aquifer near the reservation consists of the Mississippian-age Bedford Shale, Berea Sandstone, Sunbury Shale, and Cuyahoga Shale in ascending order (Reference 61).

Scioto River Glacial Outwash Aquifer

Glacial outwash sediments and riverbed alluvium that were deposited during the Quaternary Period underlie the Scioto River Valley. 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 61).

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 62).

The Scioto River outwash aquifer supplies municipal, commercial, and domestic water for the area west of the reservation (Reference 63). 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 reservation are generally limited to published maps and hydrograph data from the Ohio Department of Natural Resources, Division of Water. Such maps for Pike County and Jackson and Vinton Counties (Reference 64) 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

With respect to aquifer contamination, the two most important aquifers are the Berea Sandstone and the Gallia (References 61, 65, 66, and 67). The ability for environmental contaminants from ACP 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 off-site 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 on-site environmental contaminants to locations exhibiting downward vertical-head profiles poses the greatest potential for off-site 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 DOE reservation 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 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. An east-west trending groundwater divide that passes through the reservation characterizes groundwater flow patterns in both the Berea and Gallia. 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 on-site activities is possible, due to the downward vertical-head profile from the Gallia, off-site monitoring has not detected contaminant concentrations above background levels (Reference 60). Additionally, dissolved solids exceeding 10,000 ppm within about five miles down gradient from the reservation 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 reservation is estimated at between 2.3 and 11.7 in. per year (Reference 66). 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 DOE reservation 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 113 ft above the stage of the Scioto River, which lies about 2 miles to the west of the reservation. The Scioto River and its tributaries receive surface water and groundwater discharge from the reservation.

Geologic units controlling groundwater flow beneath the reservation 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 reservation. 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 reservation 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 1.3-13). 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 reservation. 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 reservation. The lower 10 ft of the Berea is very similar to the underlying Bedford shale (Reference65).

1.3.5.2.1 Aquifers Beneath the Site

The Gallia exhibits the highest hydraulic conductivity of the aquifers on the DOE reservation. Hydraulic conductivity values range from 0.11 to 150 feet per day (ft/d), with a mean of 3.4 ft/d (Reference 65). Groundwater flow directions in the Gallia are roughly from the center of the reservation 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

The Berea Sandstone exhibits little spatial variation in hydraulic properties. The DOE reservation means hydraulic conductivity for the Berea is 0.16 ft/d (Reference 65). 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 between August 1988 and September 1989 indicate a mean water elevation of 646.15 ft amsl with a standard deviation of 0.92 ft (Reference 66). 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 reservation, where the directions are generally toward the east and southeast.

Recharge from precipitation has been estimated to be 8.9 in. per year using the 1985 data and the Thornthwaite method (Reference 65). 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. per 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 65). 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 DOE 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 65). 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 DOE reservation 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. These discharge routes can be potential pathways for the reservation 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 68). For example, flow in the northern part of the reservation 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 separate 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 67).

1.3.6 Geology and Seismology

This section describes the geology and seismology for the Interior Low Plateaus region of southern Ohio in the vicinity of the DOE reservation. Discussions of the site and regional physiography, reservation and engineering geography, seismology, surface faulting, and liquefaction potential are provided.

1.3.6.1 Regional and Site Physiography

The DOE 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 reservation is south of the region covered by Pleistocene glaciations. However, alluvium and transported glacial sediments form a surface veneer in the mile-wide, broad valley where the reservation is located. Erosion, exposing the underlying, nearly flat-lying shale and sandstone of Mississippian and Pennsylvanian Age have maturely dissected the surrounding hills.

The reservation 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 reservation generally range from about 660 ft to 680 ft amsl, although the ground rises to about 700 ft amsl at the base of hills that border the Perimeter Road; the surrounding hills extend up to about 1,200 ft amsl. The nearby Scioto River (at about elevation 510 ft amsl) 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 amsl) 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 DOE reservation 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 reservation, 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 amsl 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 reservation. 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 reservation prior to glaciations 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 reservation valley that has subsequently been covered by lacustrine deposits of glacial origin. Only the small streams that flow through the reservation contain recent alluvium.

Unconsolidated deposits at the reservation 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.

<u>**Unconsolidated material**</u>

Fill – Fill was placed during the 1950s to develop the reservation. 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 reservation and underlie fill at the remainder of the reservation; 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, over consolidated, 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 reservation, with the base of the formation at elevation 639 ft amsl. 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 reservation.

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 reservation. 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 the DOE reservation bedrock; some streambeds contain recent alluvium. Little bedrock is exposed on the reservation 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 (Reference 18). The available data indicates that the underlying bedrock is not faulted; it has a strike of north 28° 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 DOE reservation 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 reservation 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 over consolidated 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 reservation area indicate no mineral extraction has been done beneath the reservation. 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 reservation.

The soil at the reservation 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 reservation for sanitary or process uses.

The exploratory and laboratory test data indicate that the glacial and alluvial soils are over consolidated 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 (Reference 18), south-southeast and southwest of the GDP, found groundwater between 650 ft amsl and 665 ft amsl. 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: 1 vertical or greater except for one slope; this slope is not adjacent to any structures (Reference 69). Low inclination slopes less than 20 ft in height that have soil parameters of $\phi = 10°$, $c = 1{,}000$ 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 occurs 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 DOE reservation and there have been no historical earthquake epicenters within less than 25 miles from the reservation except for two small recent events. On December 21, 2014, a magnitude 2.0 event occurred in Union Township of Pike County, approximately four miles southeast of the DOE reservation. On March 20, 2019, a magnitude 2.1 event occurred in Minford, Scioto County, approximately 12 miles southeast of the DOE reservation (Reference 70). There have been eight other earthquake epicenters within 50 miles. The maximum event had an epicenter intensity of over IV on the Modified Mercalli (MM) scale. But these events were at the reservation 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 reservation was in 1955 and had a value of only 0.005 gravity.

In the Preliminary Safety Analysis Report (Reference 15) developed for GCEP and issued in July 1980, the documented results of the studies of the historic seismicity of the area surrounding the reservation were presented. 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.

The seismic design criteria for the GCEP site was published in a DOE document, ORO-EP-120, *Seismic Design Criteria for the Gas Centrifuge Enrichment Plant - GCEP* (Reference 16) in 1980 and contained recommended design and maximum earthquake PGA values. The PGA values corresponding to these two earthquake levels were 0.04 gravity for the design earthquake and 0.15 gravity for the maximum earthquake corresponding to 72- and 1,000-year return periods, respectively. These PGA levels were selected based on judgment considering: 1) much of the information discussed in the 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) the 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 conservative.

The DBE for the primary facilities in the ACP is a 1,000-year return period earthquake, except for the X-3344 Customer Services Building which has a 10,000-year return period earthquake DBE or 0.48 gravity PGA value. Updated seismic criterion were developed specifically for the ACP and referenced in the *Summary of ACP Seismic Design Values* (Reference 29). The document summarizes the DBEs for the current site-specific return periods of 1,000 and 10,000-years. Additionally, the document includes the 100,000-year response spectra which is used to show there is adequate reserve in the connections for the X-3344 which is designed for a 10,000-year DBE. This criterion was based on earlier geotechnical investigations performed by

Engineering Consulting Services (ECS) and Fugro, Williams, Lettis and Associates (FWLA) and presented in these reports: ECS, *Final Report of Site-Specific Seismic Study* dated January 2006 (Reference 21), ECS, *Final Report of Subsurface Exploration and Geotechnical Engineering Evaluation* dated March 2006 (Reference 30), and FWLA, *Geotechnical Investigation – American Centrifuge Plant* dated June 2010 (Reference 31). Further description of seismic acceleration justification can be found in Sections 2.5.1.1 and 6.1.1.7 in the ISA Summary.

1.3.6.6 Surface Faulting

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

1.3.6.7 Liquefaction Potential

Extensive exploration and laboratory testing programs (data sets) have been completed at the DOE reservation. The associated borings and accompanying laboratory test results were used at the reservation 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 reservation, 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 reservation area due to liquefaction is unlikely.

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

	1980	1990	2000	2010	2020
Jackson County	30,592	30,230	32,641	33,225	31,600
Pike County	22,802	24,249	27,695	28,709	29,000
Ross County	65,004	69,330	73,345	78,064	76,000
Scioto County	84,545	80,327	79,195	79,499	73,730
Region of Influence	202,943	204,136	212,876	219,497	210,330
Ohio	10,797,630	10,847,115	11,353,140	11,536,504	11,574,870

Year 2020 projections based on established rates applied to 2010 census counts.

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

Recurrence Interval (Years[b])	Storm duration (hours) 0.5	1	2	3	6	12	24
	Precipitation (in.[a])						
1	0.85	1.08	1.33	1.47	1.72	1.99	2.29
2	1.03	1.31	1.62	1.79	2.09	2.43	2.79
5	1.27	1.61	1.98	2.19	2.57	2.98	3.42
10	1.48	1.88	2.33	2.57	3.01	3.49	4.01
25	1.8	2.29	2.82	3.12	3.65	4.24	4.87
50	2.09	2.66	3.28	3.62	4.24	4.92	5.66
100	2.4	3.06	3.77	4.16	4.88	5.66	6.5
10,000	3.85	4.91	6.05	6.67	7.83	9.09	10.44

[a] Values calculated based on a least-squares fit to data for 1 to 100 year recurrence interval (Reference 4)

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

	Elevation	
Recurrence interval	Meters	Feet
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
Nominal grade	204.2	670.0

[a] Estimates by U.S. Army Corps of Engineers .

[b] Estimated from records at Higby, 181.0 m (593.7 ft), assuming the flood level at the mouth of Big Beaver Creek is 11.3 m (37 ft) lower.

[c] Probable Maximum Flood calculated flow is greater than that of the estimated 10,000-year flood discharge.

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

Figure 1.3-1 Topographic Map of the Department of Energy Reservation

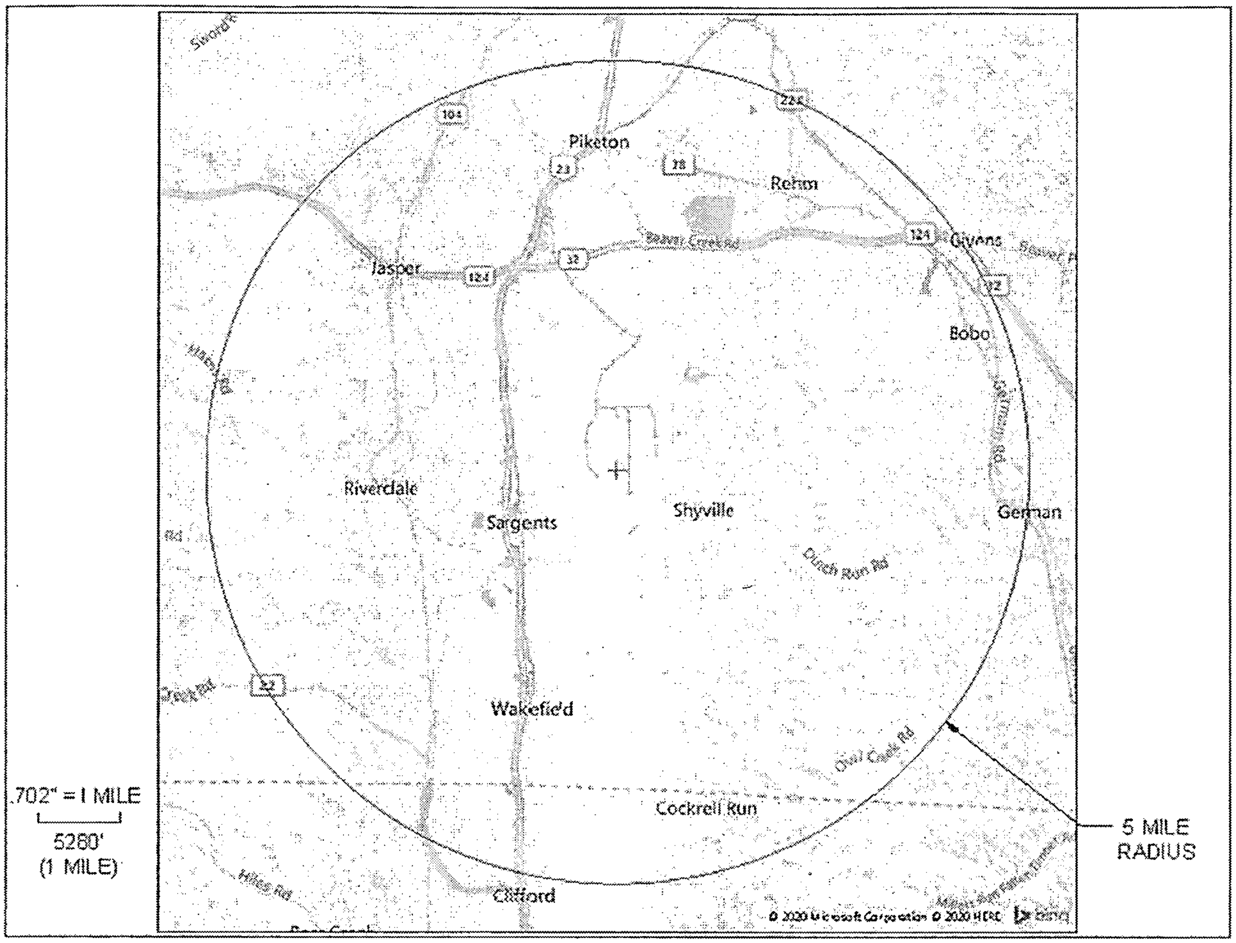

Figure 1.3-2 Area Within Five Mile Radius of the U.S. Department of Energy Reservation

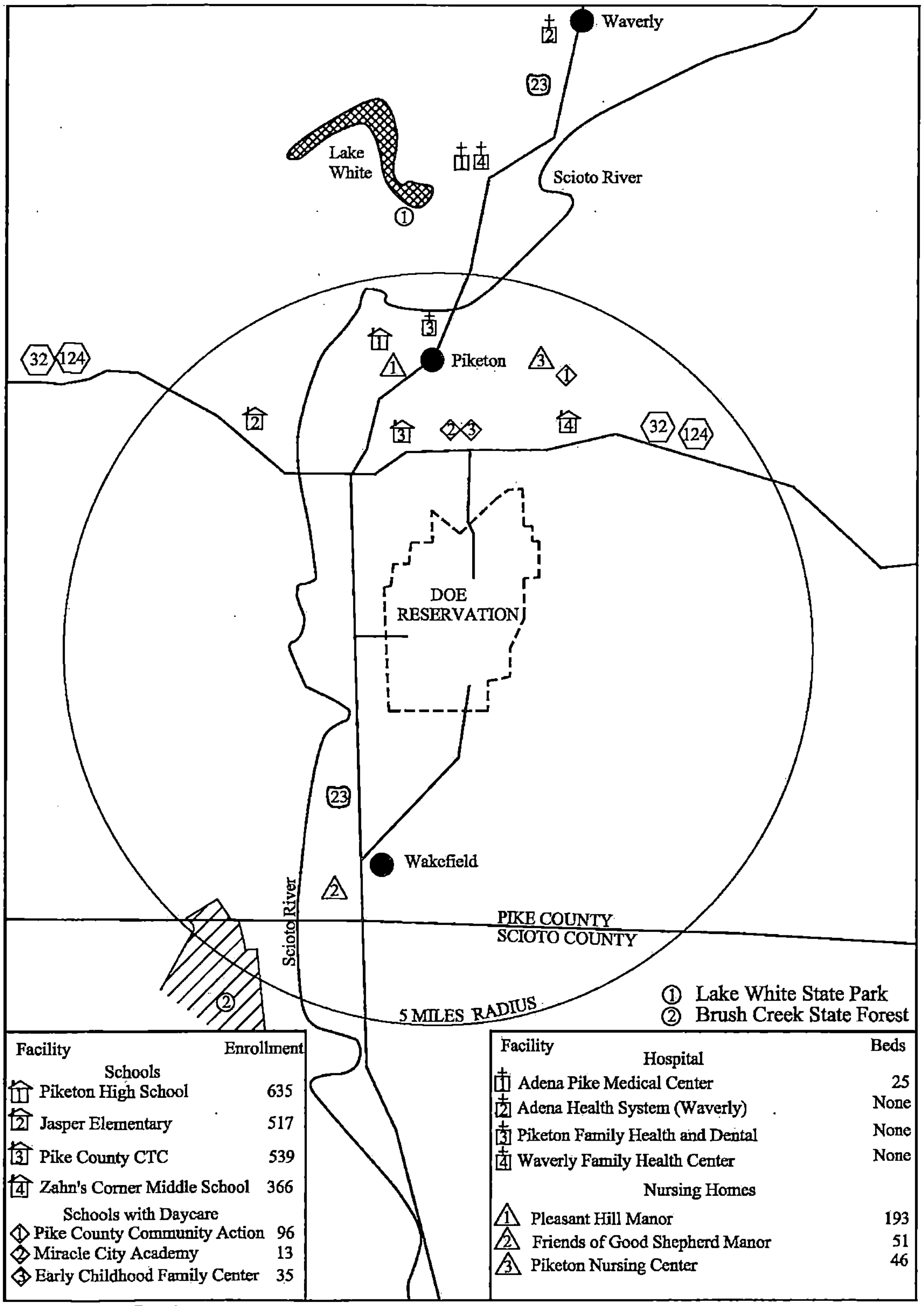

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Figure 1.3-3 Special Population Centers Within Five Miles of the U.S. Department of Energy Reservation

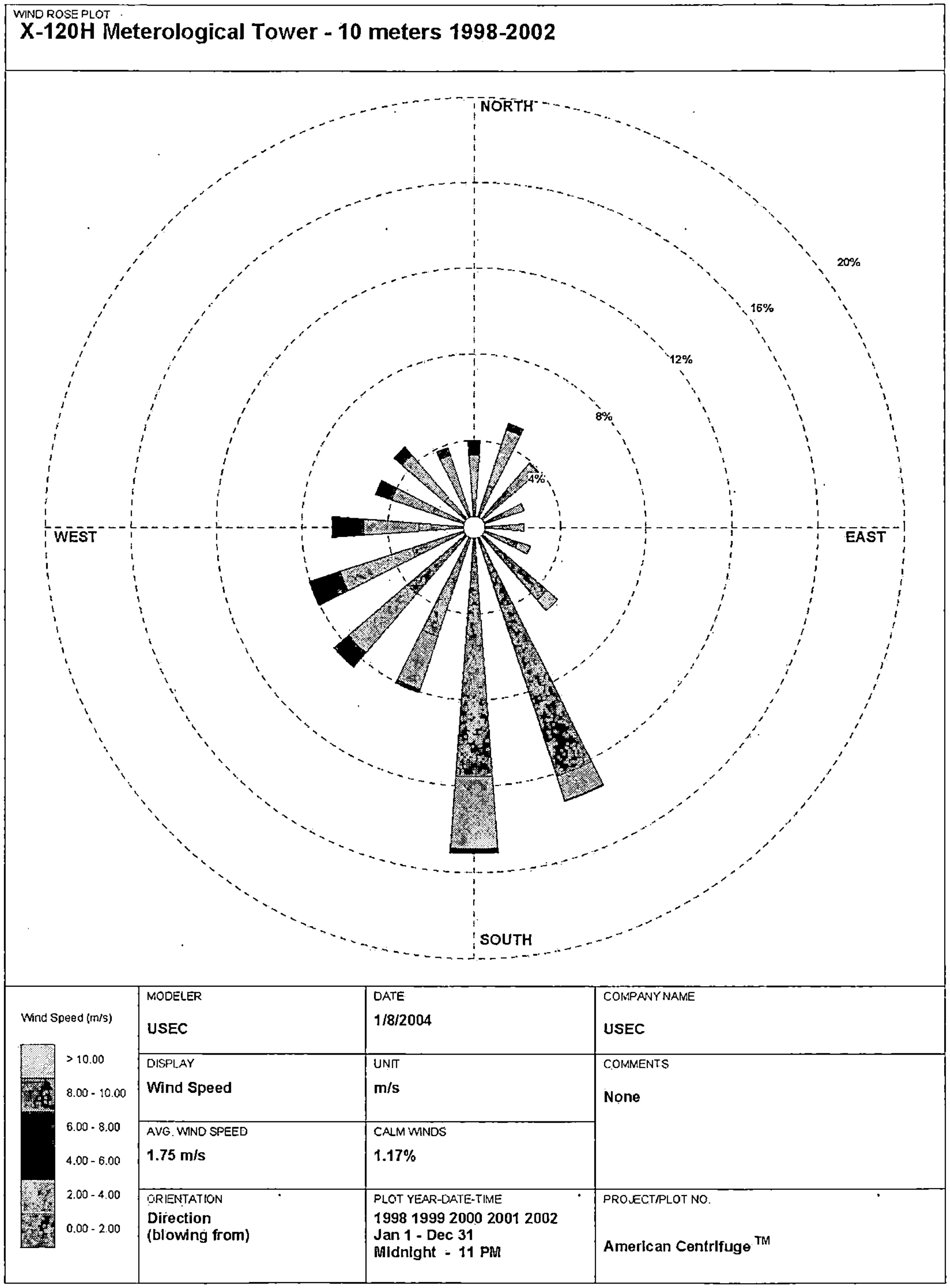

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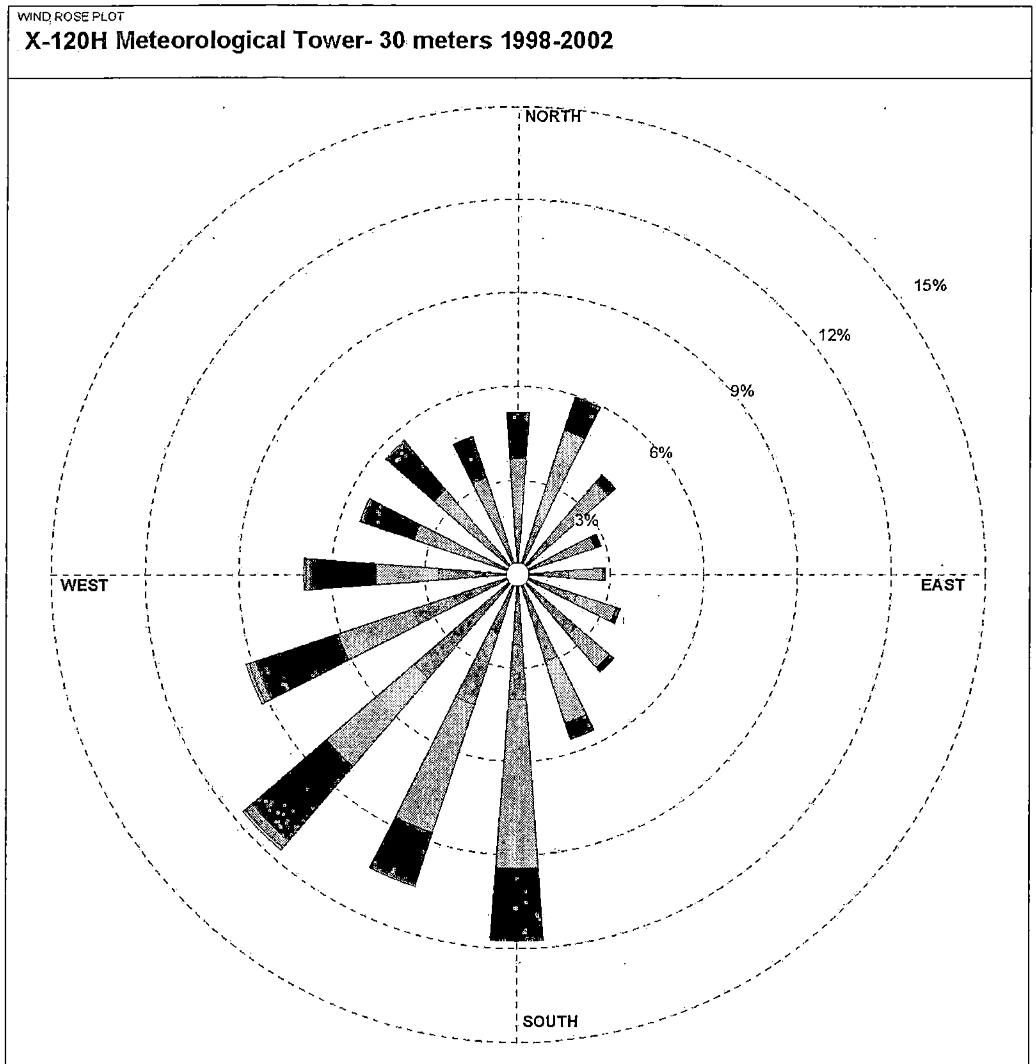
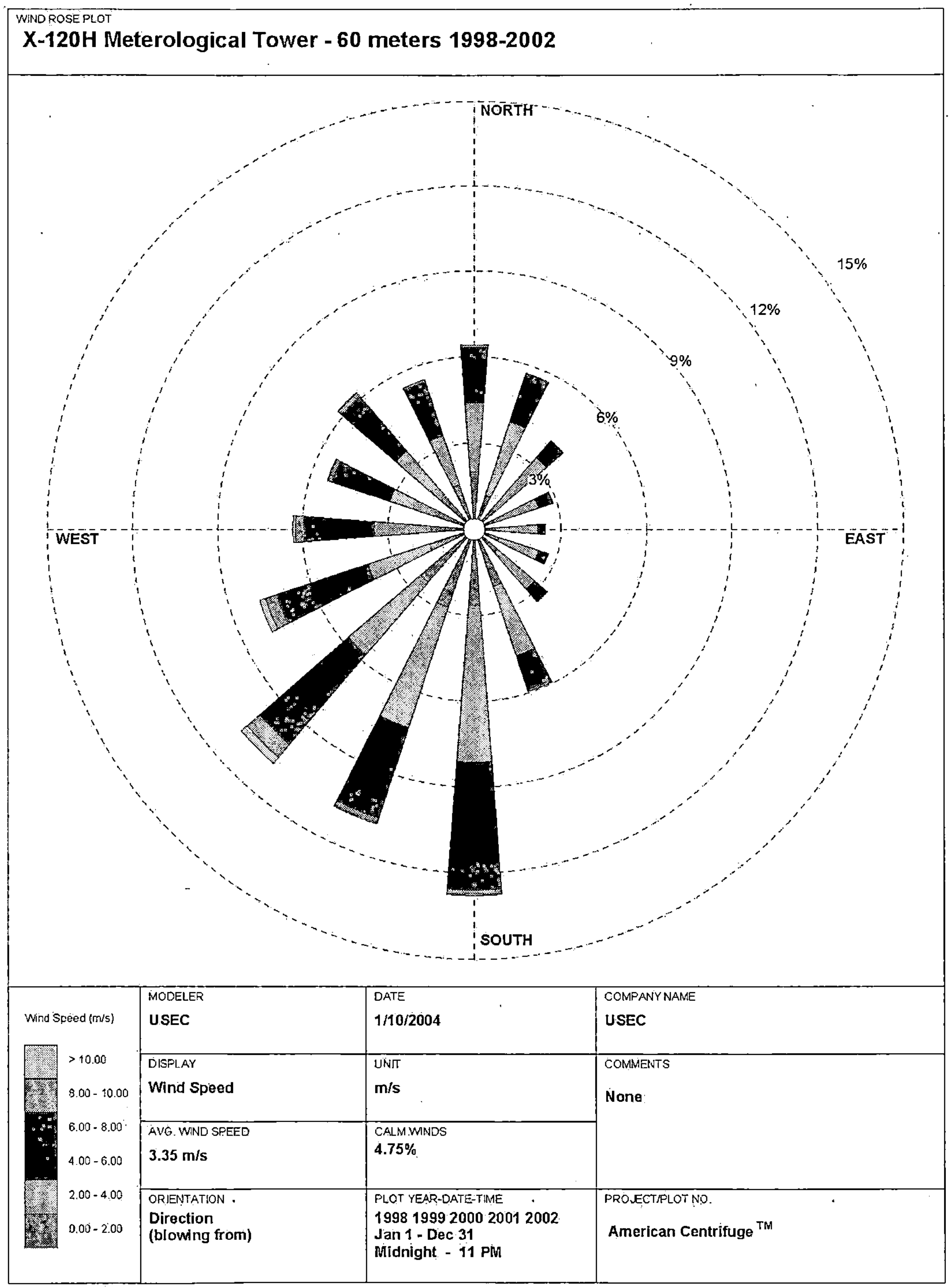
Figure 1.3-4 Comparison of Wind Roses at 10-m Level at the U.S. Department of Energy Reservation from 1998 - 2002



Wind Speed (m/s)			
	MODELER USEC	DATE 11/26/2003	COMPANY NAME USEC
> 10.00 8.00 - 10.00	DISPLAY Wind Speed	UNIT m/s	COMMENTS None
6.00 - 8.00 4.00 - 6.00	AVG. WIND SPEED 2.7 m/s	CALM WINDS 0.05%	
2.00 - 4.00 0.00 - 2.00	ORIENTATION Direction (blowing from)	PLOT YEAR-DATE-TIME 1998 1999 2000 2001 2002 Jan 1 - Dec 31 Midnight - 11 PM	PROJECT/PLOT NO. American Centrifuge ™

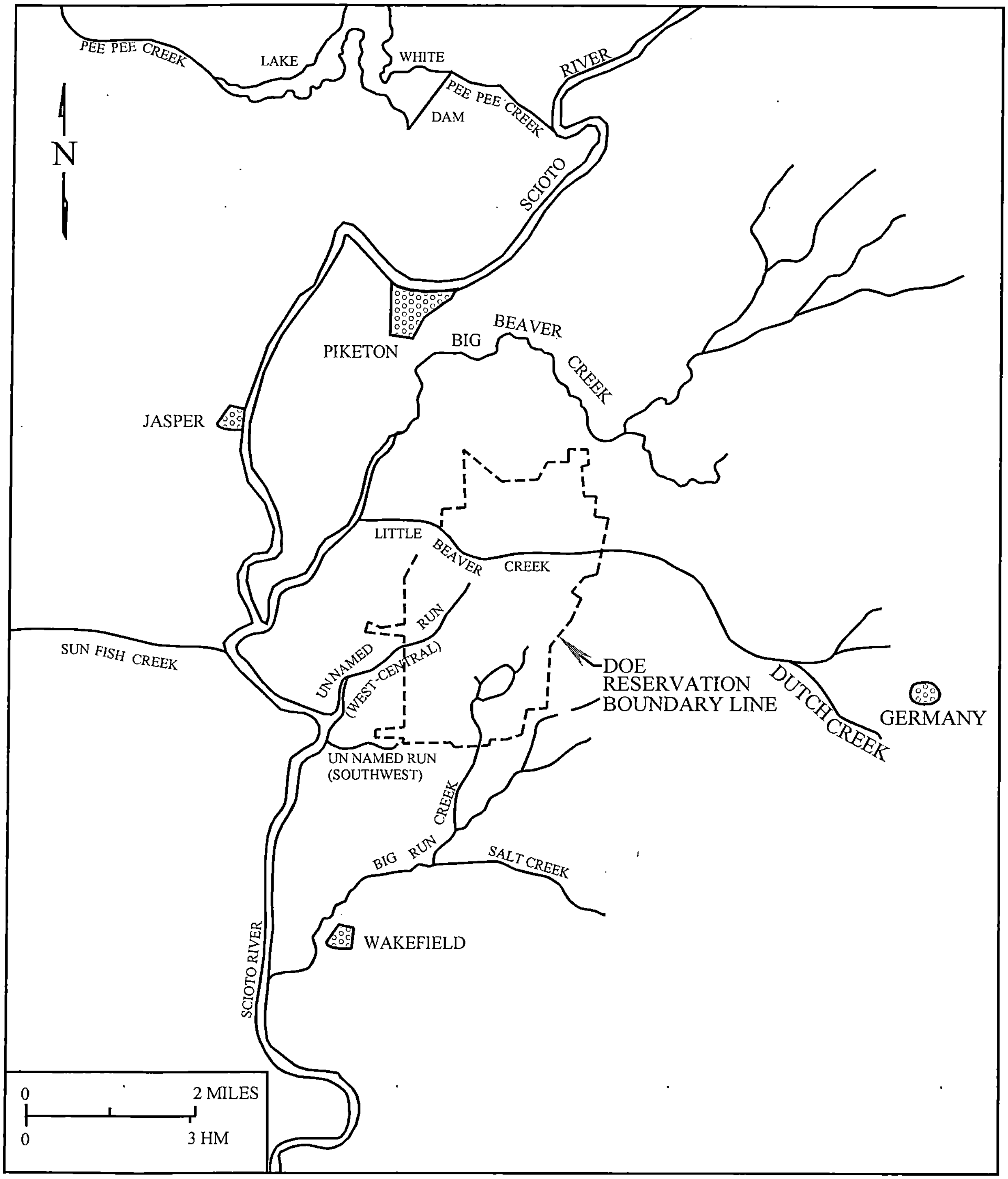
CP-058-R0

Figure 1.3-5 Comparison of Wind Roses at 30-m Level at the U.S. Department of Energy Reservation from 1998 - 2002



CP-059-R0

Figure 1.3-6 Comparison of Wind Roses at 60-m Level at the U.S. Department of Energy Reservation from 1998 - 2002



CP-038-R0

Figure 1.3-7 Location of Rivers and Creeks in the Vicinity of the U.S. Department of Energy Reservation

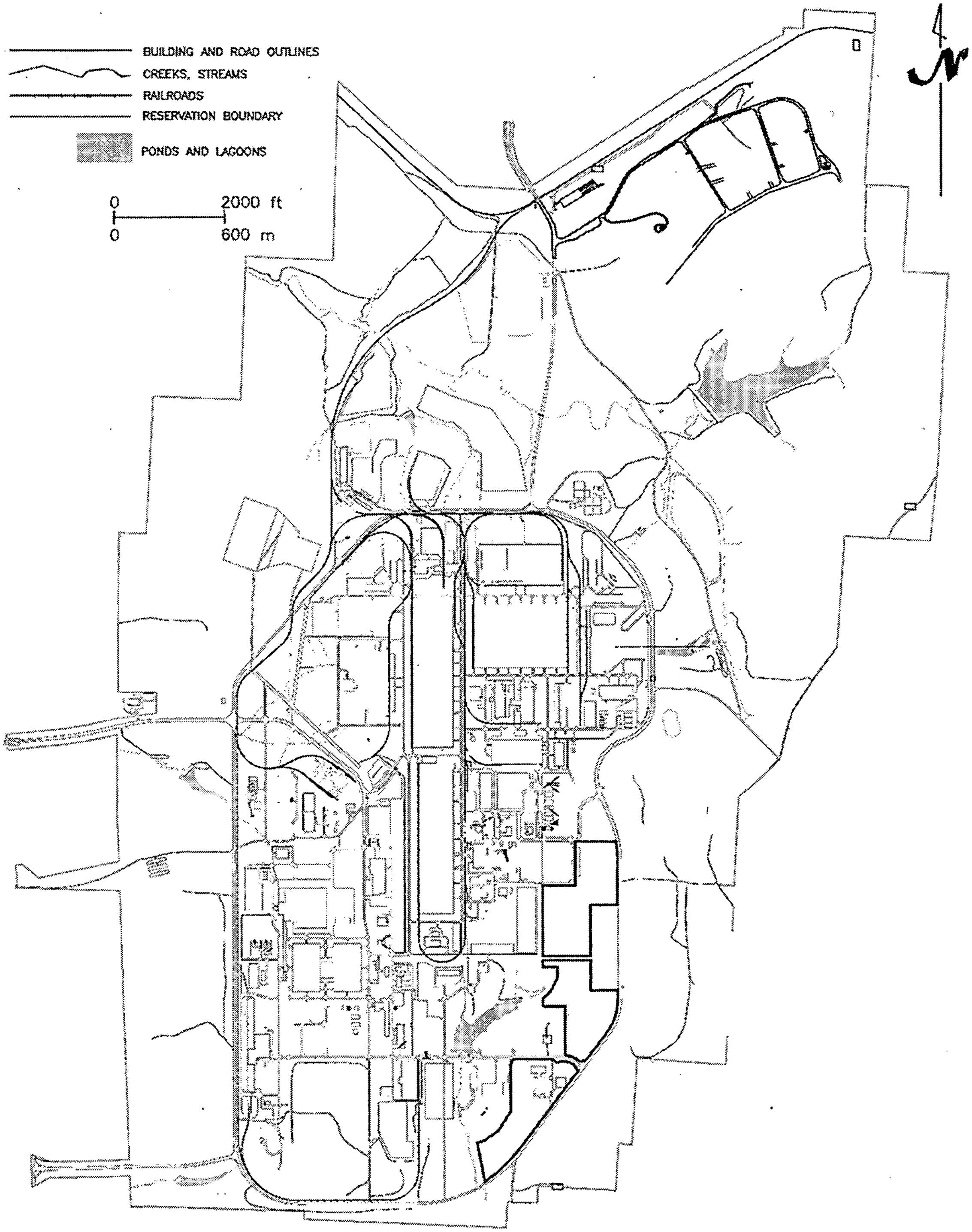


Figure 1.3-8 Ponds and Lagoons on the U.S. Department of Energy Reservation

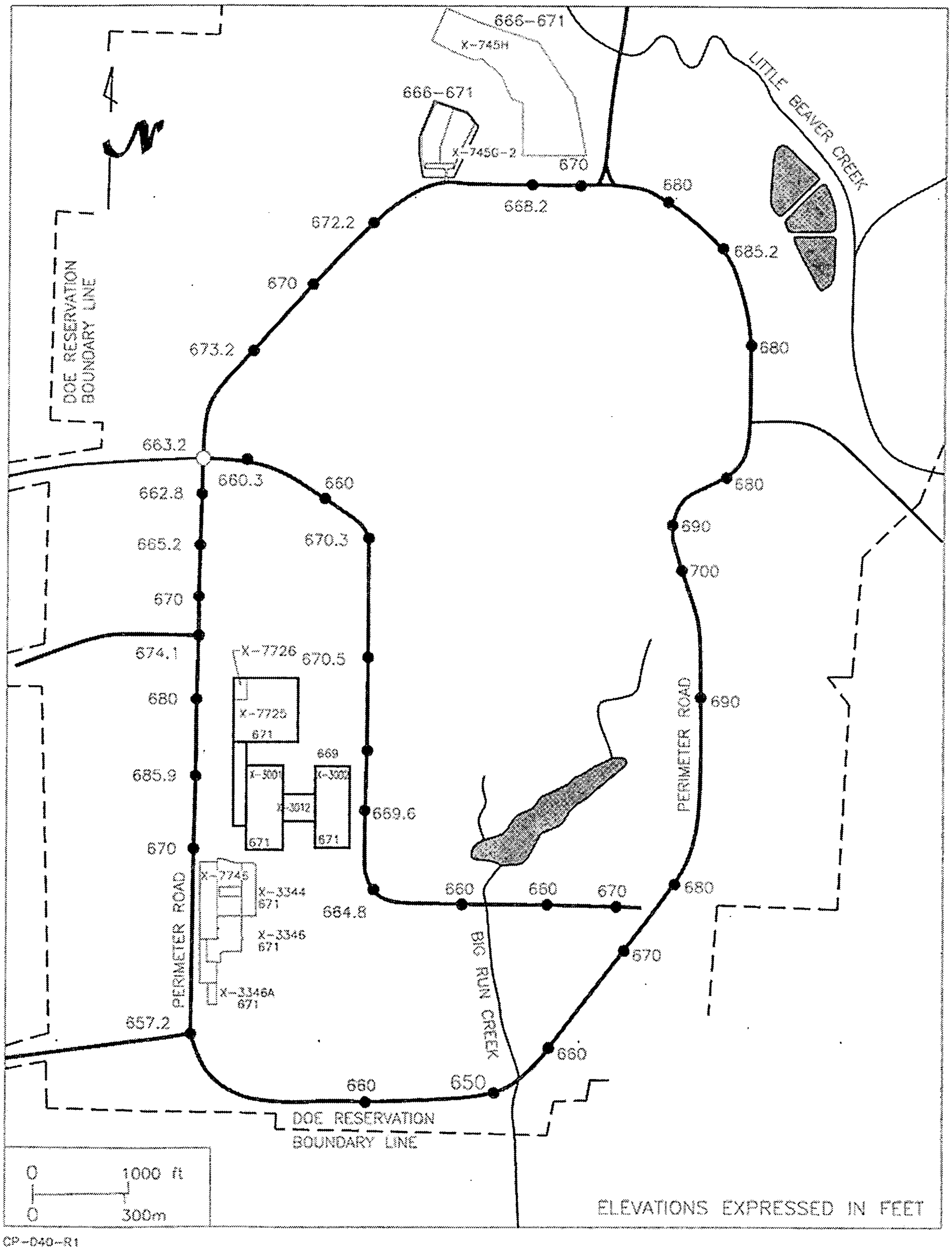


Figure 1.3-9 Elevations of Roadways and of the Surrounding Areas of Main Process Buildings

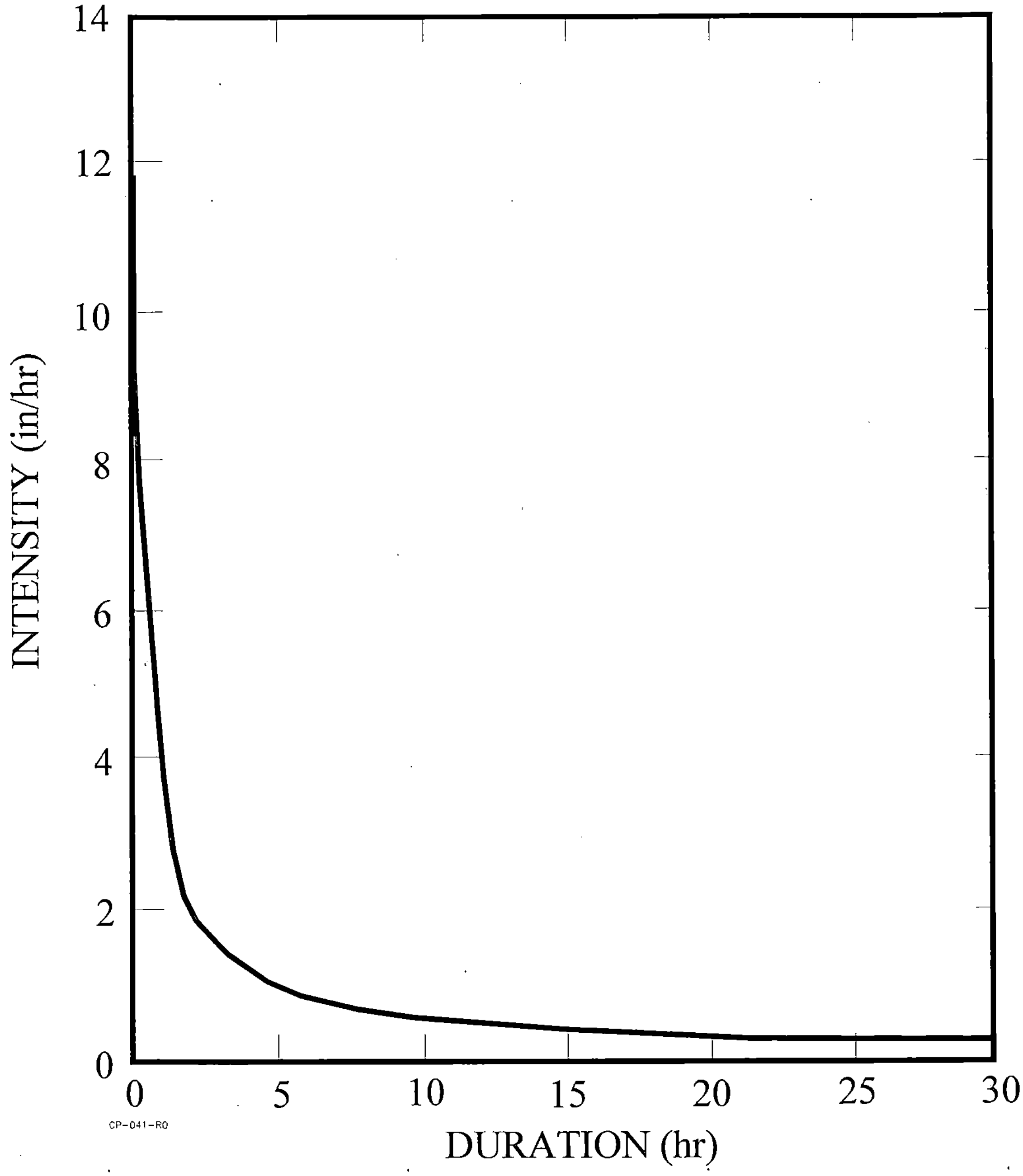


Figure 1.3-10 The 10,000-year Intensity Versus Duration Graph for Storms at U.S. Department of Energy Reservation

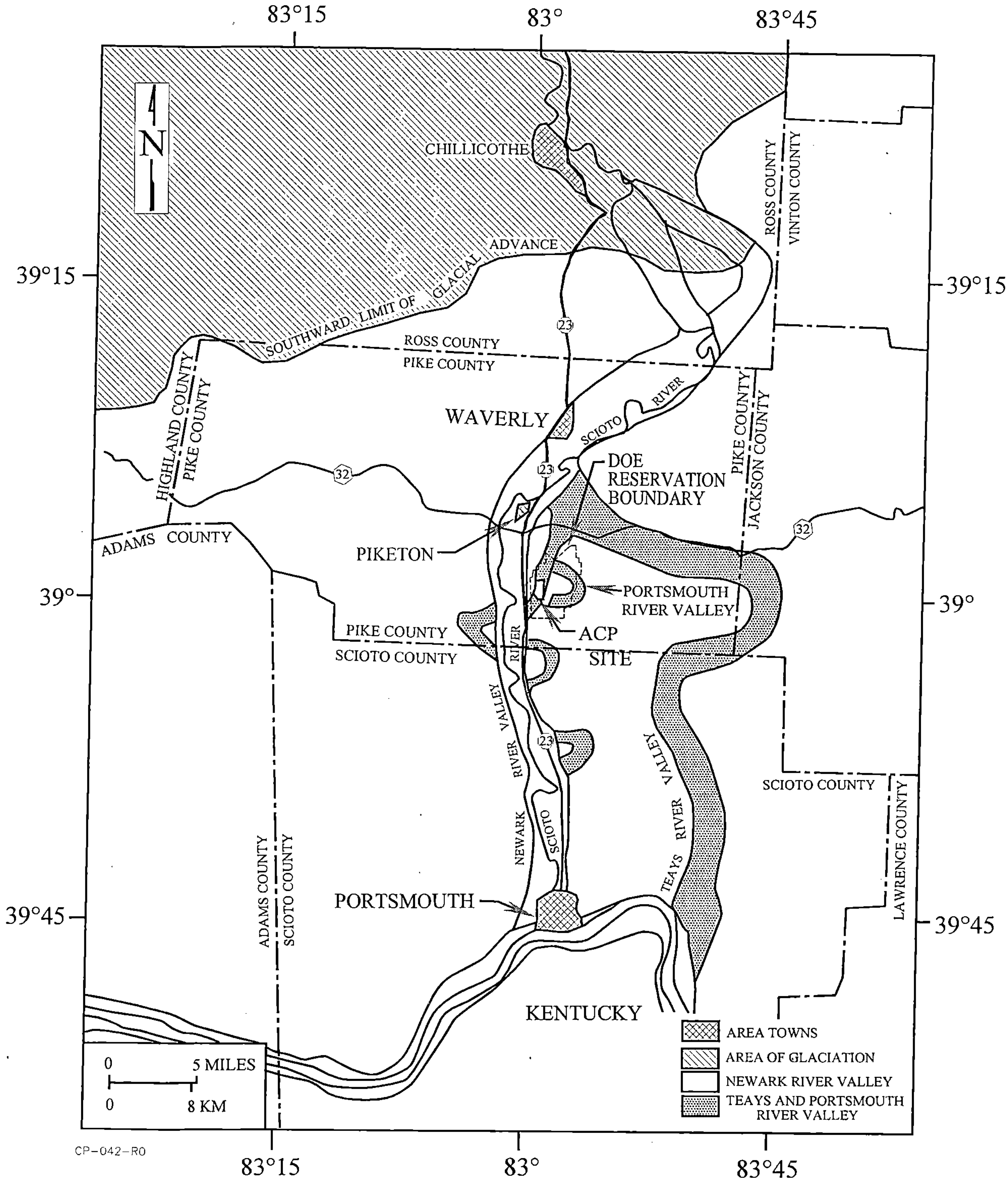


Figure 1.3-11 Location of the Ancient Newark (Modern Scioto) and Teays Valleys in the U.S. Department of Energy Reservation Vicinity

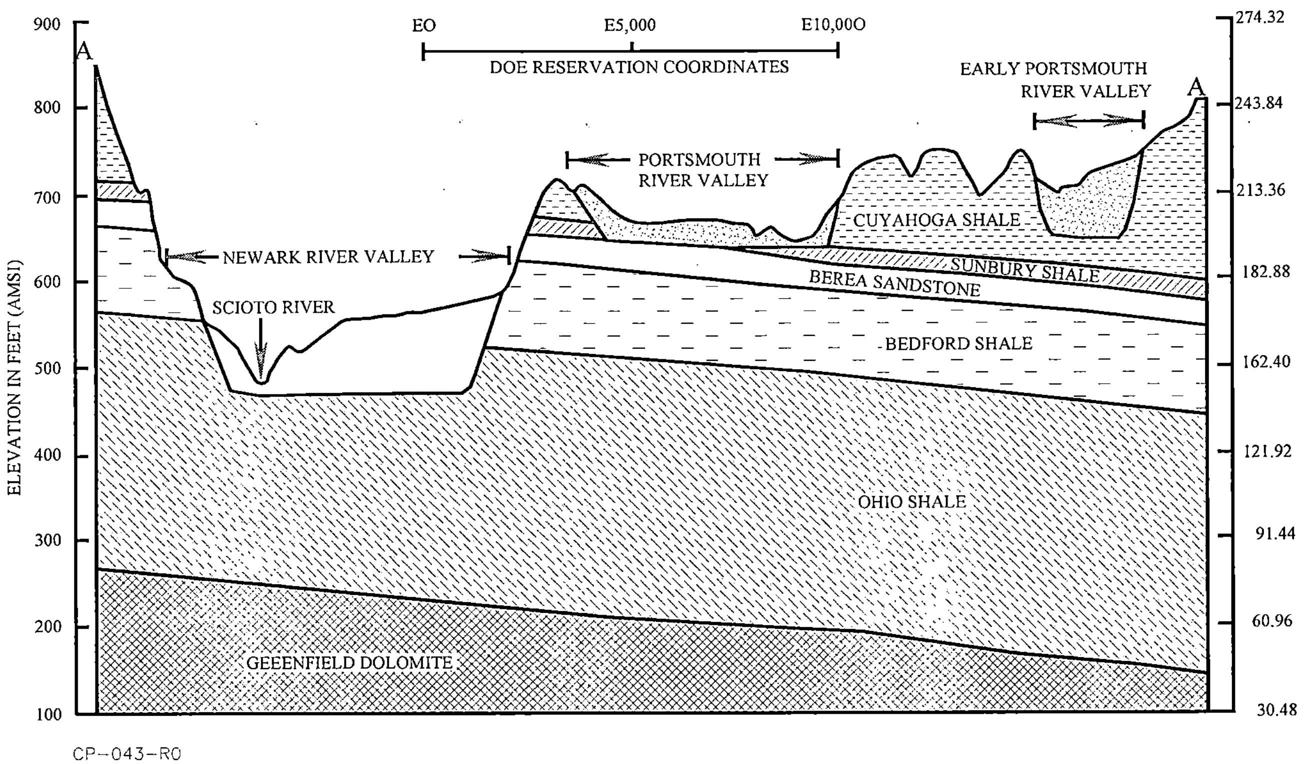


Figure 1.3-12 Geologic Cross Section in the U.S. Department of Energy Reservation Vicinity

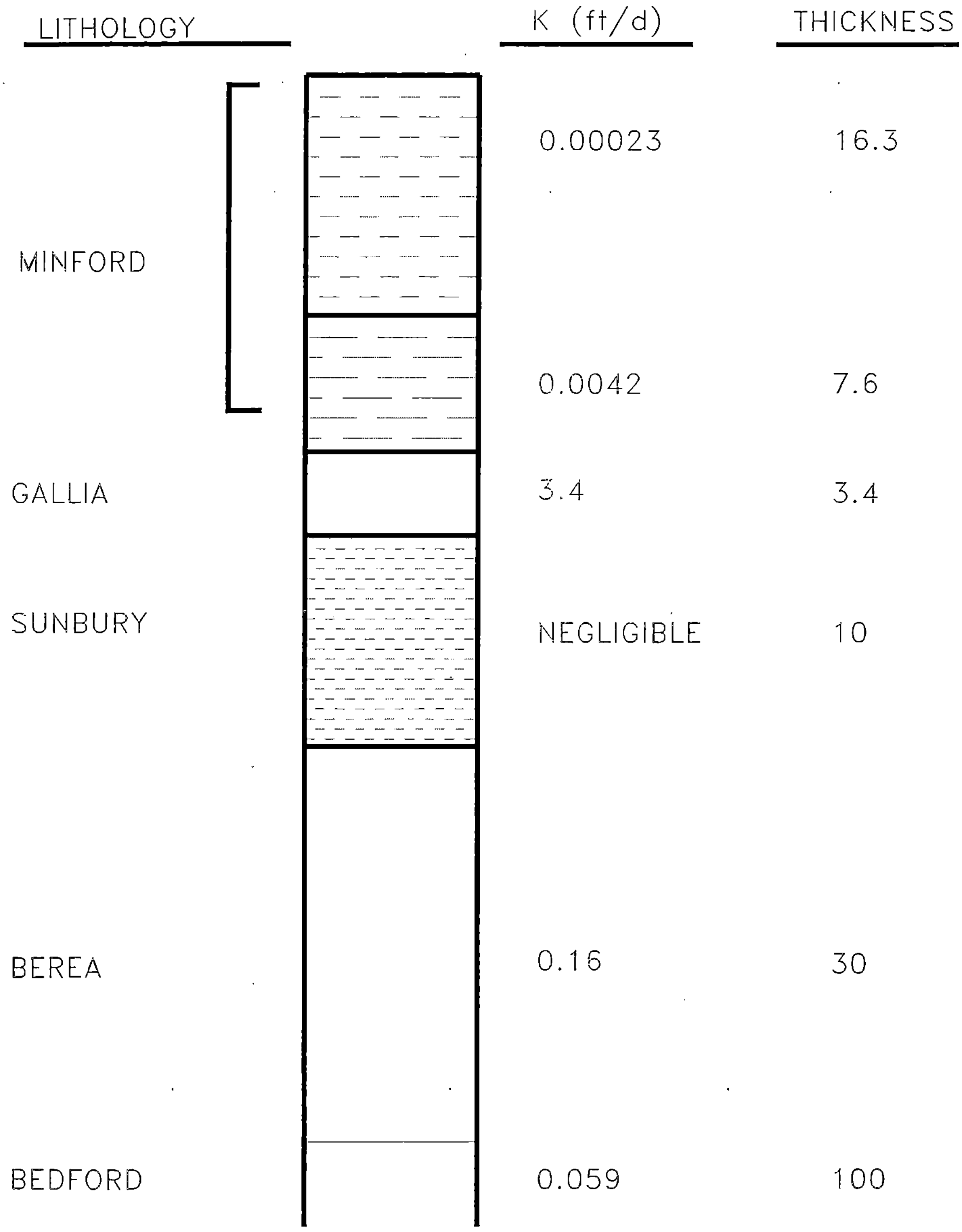


Figure 1.3-13 Geologic Column at the U.S. Department of Energy Reservation

1.4 Application Codes and Standards

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

The following sub-sections list the various industry codes and standards that have been referenced in this license application. The extent to which the Licensee satisfies the requirements of each code or standard is identified individually in the sub-sections. In the context of this section, the terms provisions and guidance are intended to refer only to the explicit requirements of each code or standard.

To establish definitive guidance for the design of the ACP, the Licensee proposed that the license be conditioned as follows:

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

The current design of the ACP does not include any items relied on for safety (IROFS) that use software, firmware, microcode, Programmable Logic Controllers, and/or any digital device, including hardware devices that implement data communication protocols.

1.4.1 American National Standards Institute/American Nuclear Society

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

  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 11.4.2.1 of this license application.

- ANSI/ANS 8.1-2014, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*

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

  Section 4.1.6 - Operations are reviewed annually; however, personnel in the operating group who are knowledgeable of the NCS requirements for their operations perform this review. 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 annually.

  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-8.3-1997, *Criticality Accident Alarm System*

  The Licensee satisfies the provision of this standard as modified by Regulatory Guide 3.71 with the following exceptions/clarifications:

  Section 1.2.5 – The primary radiation alarm system is the Criticality Accident Alarm System designed to detect a nuclear criticality and provide annunciation using audible alarms that are supplemented by visual alarms in some locations (e.g., in high-noise areas) that will alert personnel to evacuate the immediate area. ACP primary facilities that handle ^{235}U in quantities greater than 700g have Criticality Accident Alarm System coverage except the UF_6 cylinder storage yards.

  For reference to this standard, see Sections 5.4.1, 5.4.4, and 8.1.1 of this license application; Section 2.2.4 of the Emergency Plan for the American Centrifuge Plant; and Section 3.10.6 of the ISA Summary for the ACP.

- ANSI/ANS-8.19-2014, *Administrative Practices for Nuclear Criticality Safety*

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

  Section 8.6 - Operations are reviewed annually; however, personnel in the operating group who are knowledgeable of the NCS requirements for their operations perform this review. 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 biennially (every two years).

For references to this standard, see Sections 5.4.1 and 11.3.1.8 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 5.4.1, 11.3.1.1.2, 11.3.1.4, and 11.3.1.8 of this license application.

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

  The Licensee satisfies the provisions of this standard.

  For references to this standard, see Section 5.4.1 and 5.4.5 of this license application.

- ANSI/ANS-8.23-2007, *Nuclear Criticality Accident Emergency Planning and Response*

  The Licensee satisfies the provisions of this standard as modified by Regulatory Guide 3.71. Section 4.1(9) of the standard requires provision for nuclear accident dosimeters meeting ANSI N13.3-1969 (Reaffirmed 1981), "Dosimetry for Criticality Accidents." A clarification is that nuclear accident dosimeters may be used that do not necessarily comply with ANSI N13.3-1969 (R1981).

  For references to this standard, see Section 5.4.1, 5.4.4, and 8.1.1 of this license application and Section 2.2.4 of the Emergency Plan for the American Centrifuge Plant.

- ANSI/ANS-8.24-2017, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*

  The Licensee satisfies of this standard as modified by Regulatory Guide 3.71.

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

- ANSI/ANS-8.26-2007, *Criticality Safety Engineer Training and Qualification Program*

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

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

For references to this standard, see Sections 5.2.2, 5.4.1, and 11.3.1.8 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 Sections 4, 5, 6, and 7 of this standard for determining radiation protection exposure records.

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

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

  The Licensee satisfies the provisions of this standard, except for Sections 4.6 and 5.1(3).3.

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

- ANSI N14.1-2012, *Nuclear Materials - Uranium Hexafluoride - Packaging for Transport*

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

  A. Cylinders, Valves, and Plugs: Cylinders, valves, and plugs are manufactured or purchased to ANSI N14.1-2012.Previously procured and manufactured cylinders, valves, and plugs that meet previous versions of the ANSI standards or specifications in effect at the time of manufacture may be used. Alternatively, existing cylinders, valves, and plugs manufactured to previous version of the ANSI standards or specifications may be modified to meet ANSI N14.1-2012 at some point in the lifecycle due to potential issues or constraints that prohibit continued compliance with standard or specification in effect at the time of manufacture. Only cylinders, valves, and plugs of models still authorized by ANSI N14.1-2012 for manufacture may be accepted for this modification. Cylinders of this type may be subsequently transferred to the ACP.

  B. Cylinder Plugs: Use of steel or aluminum-bronze plugs in UF_6 cylinders was acceptable at the United States Enrichment Corporation GDP's for the following operations: heating, feeding, sampling, filling, transferring between cylinders, and onsite transport and storage. Therefore, these cylinders with these types of plugs may be subsequently transferred to the ACP.

For the reference to this standard, see Section 1.1.5.5.5 of this license application; Sections 2.2.3 (including subsections), 3.5.5, 3.6.4.1, and 3.7.4 (including subsections) of the ISA Summary for the ACP; and Sections 7.3.4.4, 7.3.6.4.3.1, 7.3.6.7.1.1, and 7.3.6.7.3.1, Appendix E of Addendum 1 of the ISA Summary.

1.4.3 American National Standards Institute/American Society of Mechanical Engineers

- ANSI/ASME NQA-1-2008 and NQA-1a-2009 Addenda, *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 ASME NQA-1-2008 with NQA-1a-2009 addenda, Part I, Introduction, Section 400 *Terms and Definitions*.

  B. Indoctrination and training satisfies the provisions of ASME NQA-1-2008, Part I, Requirement 2, Section 200 *Indoctrination and Training* and Section 500 *Records*.

  C. Personnel performing inspection and testing, as well as QA audit personnel, meet the requirements of ASME NQA-1-2008, Part I, Requirement 2, Section 300 *Qualification Requirements* and Section 400 *Records of Qualification*.

  D. Design controls that consist of computer programs are developed, validated, and managed in accordance with ASME NQA-1-2008 with the NQA-1a-2009 addenda, Part I, Requirement 3, Design Control, Section 800, Requirement 11 *Test Control* and Part II, Subpart 2.7 *Quality Assurance Requirements for Company Software for Nuclear Facility Applications*.

  E. Methods of design verification satisfy the provisions of ASME NQA-1-2008, Part I, Requirement 3, Section 501 *Methods*.
  F. Computer Program Testing is performed in accordance with ASME NQA-1-2008 with the NQA-1a-2009 addenda, Part I, Requirement 11, *Test Control*.
  G. Lifetime records are defined in accordance with ASME NQA-1-2008, Part I, Requirement 17, Section 401 *Lifetime Records*.

  H. Hard copy or microfilm storage facilities satisfies the guidance of ASME NQA-1-2008, Part I, Requirement 17, Section 600 *Storage*.

For the references to this standard, see Section 11.5.1 of this license application and Sections 2.0, 3.0, and 11.0 of the QAPD for the ACP.

1.4.4 American Society of Mechanical Engineers

- ASME Boiler and Pressure Vessel Code Section VIII, *Pressure Vessels*, 2004

  Autoclaves providing containment to minimize the potential for release of licensed material are designed, constructed, and installed in accordance with this standard.

  For the references to this standard, see Sections 3.6.4.1 and 7.3.4.16 of the ISA Summary.

- ASME B31.3, *Process Piping*, 2018

  Piping providing containment to minimize the potential for release of licensed material is designed, constructed, and installed in accordance with this standard.

  For the references to this standard, see Sections 3.6.2.3, 3.6.2.4.1, and 3.6.2.5 of the ISA Summary.

- 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 and Section 3.8.2.2 and 3.16 of the ISA Summary for the ACP.

- 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 American Society for Testing and Materials

- ASTM C787, *Standard Specification for Uranium Hexafluoride for Enrichment*, 2015

  The Licensee will satisfy the provisions of this standard. All other uranium that does not meet the requirements of ASTM - C787 for reprocessed UF_6 may be accepted for storage and subsequent dispositioning, but will not be introduced to the enrichment process, with the exception of small amounts (e.g., 50 pounds UF_6) associated with sampling, sub-sampling, and analyses required to establish receiver's values.

  For the reference to this standard, see Tables 1.2-1 and 1.2-2 of this license application.

- ASTM C996, *Standard Specification for Uranium Hexafluoride Enriched to Less than 5 Percent U-235*, 2015

  The Licensee will satisfy the provisions of this standard. All other uranium that does not meet the requirements of ASTM – C996 for reprocessed UF_6 may be accepted for storage and subsequent dispositioning, but will not be introduced to the enrichment process, with the exception of small amounts (e.g., 50 pounds UF_6) associated with sampling, sub-sampling, and analyses required to establish receiver's values.

  For the reference to this standard, see Tables 1.2-1 and 1.2-2 of this license application.

- ASTM C1052, *Standard Practice for Bulk Sampling of Liquid Uranium Hexafluoride*, 2014

  The Licensee will satisfy the provisions of this standard.

  For the reference to this standard, see Section 1.1.5.5.5 of this license application and Section 3.5.5 of the ISA Summary.

1.4.6 National Fire Protection Association

- NFPA 10-2018, *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 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 Section 7.4.3 and Table 7.1-1 of this license application.

- NFPA 13-2019, *Standard for the Installation of Sprinkler Systems*

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

  > Existing suppression systems are maintained in accordance with the applicable codes and standards enforced at the time of construction and installation. The provisions of the standard in place at the time of construction and installation were used as guidance for the design and installation of wet and dry pipe automatic sprinkler systems. In addition, ACP facilities meet the definition of Ordinary Hazard Occupancies (Group 1) as stated in this standard and the fire protection systems meet or exceed the sprinkler discharge requirements for this type of occupancy. 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 7.3.1 and Table 7.1-1 of this license application and Section 3.10.3 of the ISAS for the ACP.

- NFPA 15-2017, *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 7.3.1 and Table 7.1-1 of this license application.

- NFPA 25-2002, *Standard for Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*

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

For the reference to this standard, see Section 7.1.2 and Table 7.1-1 of this license application and Sections 2.2.6 and 3.8.1.1 of the ISA Summary for the ACP.

- NFPA 30-2018, *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 Section 7.3 and Table 7.1-1 of this license application.

- NFPA 51B-2019, *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 7.1.1, 7.1.2, and Table 7.1-1 of this license application.

- NFPA 55-2020, *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 7.1.1, 7.3, and Table 7.1-1 of this license application.

- NFPA 70-2005, *National Electrical Code*

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

  For the reference to this standard, see Section 7.3 and Table 7.1-1 of this license application and Section 2.6.7 of the ISA Summary for the ACP.

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

  This NFPA standard was used as guidance for the installation of the fire alarm systems.

  For the reference to this standard, see Section 7.3.2 and Table 7.1-1 of this license application.

- NFPA 75-2003, *Standard for the Protection of Electronic Computer/Data Processing Equipment*

  This NFPA standard was used as guidance for the protection of the computer systems.

  For the reference to this standard, see Chapter 7 of this license application.

- NFPA 80-1999, *Standard for Fire Doors and Fire Windows*

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

  For the reference to this standard, see Chapter 7 of this license application.

- NFPA 101-2018, *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 Chapter 7 of this license application.

- NFPA 220-1999, *Standard on Types of Building Construction*

  The Licensee uses the provisions of this standard as guidance for the review of building construction.

  For the reference to this standard, see Table 7.1-1 of this license application.

- NFPA 241-2019, *Standard Safeguarding Construction, Alteration, and Demolition Operations*

  The Licensee uses the provisions of this standard as guidance for the review of construction activities.

  For the reference to this standard, see Section 7.1.1 and Table 7.1-1 of this license application.

- NFPA 801-2020, *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 7.0 of this license application.

  For the reference to this standard, see Section 7.0 and Table 7.1-1 of this license application.

1.4.7 Section Reserved For Future Use

1.4.8 Institute of Electrical and Electronics Engineers

Several of the Institute of Electrical and Electronics Engineers (IEEE) standards identified in this section include the term "Class 1E." The Licensee is taking exception to utilizing the term "Class 1E." The term utilized by the Licensee for items relied on for safety, per 10 CFR Part 70, is "IROFS." IROFS quality levels (i.e., QL-1 or QL-2) are established and defined in Section 2.0 of the QAPD. The IROFS, including their quality class, are based on the analyzed, credible conditions identified in the ISA. IROFS (and non-IROFS that may directly affect the safety function of an IROFS) will be designed, procured, maintained and documented in accordance with the requirements of the "Configuration Management Program" included in Chapter 11.0 of this license application.

- ANSI/IEEE 336-2010, *ANSI/IEEE Standard Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities*

  The Licensee commits to periodic inspections and testing of items relied on for safety will be in accordance with Clause 7.

  For the reference to this standard see Sections 2.6.4 and 2.6.8 of the ISA Summary for the ACP.

- IEEE 338-1987 *Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems*

  The Licensee commits to utilizing IEEE 338 Sections 1 (Scope), 2 (Definitions), 4 (Basis), and 5 (Design Requirements); and portions of Sections 3 (References) and 6 (Testing Program Requirements).

  The Licensee takes exception to portions of the contents of IEEE 338 Sections 3 and 6 and Annex A for the following reasons:

Section 3	The ACP operations procedures will govern plant operations in lieu of ANSI/ANS 3.2-1982.
Section 3	In Section 3 (References) the Licensee commits to only the applicable portions of the IEEE Standards 7-4.3.2 and IEEE 603.
Section 6.1 (11)	The ACP operations procedures will govern plant operations in lieu of ANSI/ANS 3.2-1982.

  Note - Annex A provides only "informative" references.

  For the reference to this standard see Sections 2.6.4 and 2.6.7 of the ISA Summary for the ACP.

- IEEE 7-4.3.2-2003, *Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations*

  The Licensee commits to utilizing IEEE 7-4.3.2 Clauses 1 (Scope), 3 (Definitions) and 7 (Execute Features) and portions of Clauses 5 (Safety System Criteria), 6 (Sense and Command Features), and 8 (Power Source Requirements).

  The Licensee takes exception to IEEE 7-4.3.2 Clauses 2 (References), 4 (Safety System Design Basis), and Annexes A through H. These areas are not considered to be applicable or necessary due to their nuclear reactor content and redundancy with other IEEE standards and the Licensee's ISA. Annexes A through H provide only "informative" details and references. The Licensee also takes exception to the contents of IEEE 7-4.3.2 Clause 5 for the following reasons:

Sections 5.3 and 5.3.1	The Licensee commits to ASME NQA-1-2008 with NQA-1a-2009 addenda Part I, Requirement 11 and Part II, Subpart 2.7 as defined in Section 1.4.3 of this license application.
Section 5.3.2	The Licensee does not intend to qualify existing commercial computers.
Section 5.15	Reliability analysis methods and calculations are as specified in the ISA for the ACP.

  For the reference to this standard see Section 2.6.4 of the ISA Summary for the ACP.

- IEEE 308-2001, *Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations*

  The Licensee commits to utilizing IEEE 308 Section 3 (Definitions) and portions of Sections 1 (Overview), 4 (Principle Design Criteria), 5 (Supplemental Design Criteria), 6 (Surveillance and Test Requirements), and 8 (Documentation).

  The Licensee takes exception to IEEE 308 Sections 2 (References), and portions of Sections 1 (Overview), 4 (Principle Design Criteria), 5 (Supplemental Design Criteria), 6 (Surveillance and Test Requirements), and 8 (Documentation) for the following reasons:

Section 1	Figure 1 is not applicable to the ACP. The Licensee will provide reliable electrical power to all IROFS that require electrical power to function during postulated events analyzed in the ISA. Back-up power is required only as needed to provide the reliability of the IROFS as credited in the ISA. Note that IROFS that fail safe on loss of power do not require back-up power systems.
Section 2	The ACP does not commit to all of the standards listed in this section.

Section 4.2 Figure 3 is not applicable to the ACP. The Licensee will provide reliable electrical power to all IROFS that require electrical power to function during postulated events analyzed in the ISA. Back-up power is required only as needed to provide the reliability of the IROFS as credited in the ISA. Note that IROFS that fail safe on loss of power do not require back-up power systems.

Section 4.7 Documents will be identified and controlled in accordance with Sections 6.0 and 17.0 of the QAPD and plant procedures.

Sections 4.10 and 5.2.1 These Sections are not applicable to the ACP as written and are modified as follows: A back-up power supply may be utilized to provide reliable power to an IROFS that requires electrical power to function during postulated events analyzed in the ISA. The power circuits from the back-up power supply to the IROFS will be independent and redundant if necessary to provide the reliability of the IROFS as credited in the ISA. The control circuits from the control room to the IROFS will also be independent and redundant if necessary to provide the reliability of the IROFS as credited in the ISA.

Section 4.11 A non-IROFS load that needs reliable standby power may be connected to an IROFS power system in accordance with portions of Figure 3 and IEEE 384.

Sections 5.2.4 and 5.3.1 These Sections are not applicable to the ACP. The ACP will follow applicable portions of IEEE 446 for guidance related to standby power supplies and DC power systems.

Section 5.3.3.6 Battery systems for IROFS that are not failsafe will be tested in accordance with approved ACP maintenance procedures.

Section 6.1 The "illustrative" continuous monitoring surveillance methods listed in Table 3 are optional (i.e., surveillance monitoring by a computer is not mandatory).

Section 7 This section does not apply to a uranium enrichment facility.

Section 8.1 The ACP does not commit to performing the studies listed as Items a through g; applicable studies will be conducted and documented to demonstrate the adequacy of IROFS and associated support systems.

The ACP electrical IROFS systems will utilize commercial-grade equipment approved or rated by nationally-recognized industry standards and reputable organizations such as IEEE, Underwriters Laboratory Inc. (UL), Factory Mutual (FM), NFPA, and

National Electrical Manufacturers Association (NEMA). Procurement and installation will be in accordance with the QAPD.

For the reference to this standard see Sections 2.6.4 and 2.6.7 of the ISA Summary for the ACP.

- IEEE 323-2003, *Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*

The Licensee commits to IEEE 323 Clauses 1 (Scope), 3 (Definitions), 4 (Principles), and 7 (Documentation).

The Licensee takes exception to IEEE 323 Clause 2 (References), 5 (Methods), 6 (Program), and Annex A. Annex A provides only "informative" references (37), whereas, only certain portions of two IEEE standards (7-4.3.2 and 603) listed in Clause 2 (References) are applicable to the ACP.

For the reference to this standard see Sections 2.6.4 and 2.6.7 of the ISA Summary for the ACP.

Per Section 4.1, "For equipment located in a mild environment for meeting its functional requirements during normal environmental conditions and anticipated operational occurrences, the requirements shall be specified in the design/purchase specifications. A qualified life is not required for equipment located in a mild environment and which has no significant aging mechanisms." For purposes of the ACP, the equipment will be located in a mild environment in which no significant radiation exposure or aging mechanisms are identified or expected. The accident conditions anticipated at the ACP are mild in nature. The worst conditions are due to fire scenarios which can produce high temperature, subsequent water spray exposure from the fire suppression system, and exposure to UF_6 due to a release.

Therefore, the Licensee will not classify any equipment as Class 1E in accordance with Sections 5 and 6, but will include the other applicable requirements identified in the IEEE standards, i.e., design control (additional design package rigor, equipment specifications, critical design characteristics, QC inspection criteria, vendor testing requirements, special equipment storage and handling requirements), quality control, post maintenance testing, preventive maintenance/testing, surveillances and documentation control/retention.

The primary equipment that is required to fulfill the IROFS function, including necessary support system components back to the point of redundancy, is considered to be part of the IROFS boundary. All IROFS boundary components will be designed, installed and maintained to the applicable IEEE requirements identified and committed to above and in accordance with the QAPD. In addition to meeting the above requirements, the ACP electrical IROFS systems will utilize commercial-grade equipment approved or rated by nationally recognized industry standards and reputable organizations such as IEEE, UL, FM, NFPA, and NEMA.

- IEEE 379-2000, *Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems*

  The Licensee commits to utilizing IEEE 379 Sections 1 (Overview), 3 (Definitions), 5 (Requirements), and 6 (Design Analysis), and portions of Section 4 (Single-Failure Criterion). Applicable portions of IEEE 379 will be used as a guideline for the design of IROFS systems since this standard supplements IEEE 603 by providing guidance in the application of the single-failure criterion for safety systems in nuclear power stations.

  The Licensee takes exception to the contents of IEEE 379 Sections 2 and 4 and Annex A. The exceptions that the Licensee takes to the contents of IEEE 379 are:

  Section 2 The ACP does not commit to all of the standards listed in this section.

  Section 4 These Sections are not applicable to the ACP as written and are modified as follows: a back-up power system may be utilized to provide reliable power to an IROFS that requires electrical power to function during postulated events analyzed in the ISA. The power circuits from the back-up power system to the IROFS will be independent and redundant if necessary to provide the reliability of the IROFS as credited in the ISA. The control circuits from the control room to the IROFS will also be independent and redundant if necessary to provide the reliability of the IROFS as credited in the ISA.

  Annex A provides only "informative" references.

  For the reference to this standard see Sections 2.6.4 and 2.6.7 of the ISA Summary for the ACP.

- IEEE 384-1992, *Standard Criteria for Independence of Class 1E Equipment and Circuits*

  The Licensee commits to utilizing IEEE 384 Clauses 1 (Scope), 2 (Purpose), 4 (Definitions), 5 (Independence Criteria), 6 (Separation Criteria), and 7 (Specific Isolation Criteria). Applicable portions of IEEE 384 will be used as a guideline for the design of IROFS systems since this standard supplements IEEE 603 by providing guidance criteria for implementation of the independence requirements for Class 1E systems.

  The Licensee takes exception to the contents of IEEE 384 Clause 3 and Annex A. The Licensee does not commit to all the standards listed in Clause 3. Annex A provides only "informative" references.

The ACP electrical IROFS systems will utilize commercial-grade equipment approved or rated by nationally recognized industry standards and reputable organizations such as IEEE, UL, FM, NFPA, and NEMA. Procurement and installation will be in accordance with the QAPD.

For the reference to this standard see Sections 2.6.4 and 2.6.7 of the ISA Summary for the ACP.

- IEEE 446-1995, *Recommended Practice for Emergency and Standby Power Systems for Industrial and Commercial Applications*

  The Licensee commits to utilizing IEEE 446 Clauses 1 (Scope) and 2 (Definitions) and portions of Clauses 6 (Protection), 7 (Grounding), 8 (Maintenance), and 10 (Reliability).

  The Licensee takes exception to the contents of IEEE 446 Clauses 3, 4, 5, and 9. These clauses are not considered to be applicable or necessary due to their content and/or redundancy with other IEEE standards and NFPA 70 *National Electrical Code*. In addition, the Licensee takes exception to portions of IEEE 446 Clauses 6, 7, 8, and 10 for the following reasons:

Section 6.11	The Licensee does not commit to all of the standards listed in this section.
Section 7.14	The Licensee does not commit to all of the standards listed in this section.
Section 8.1.3	Maintenance personnel will receive training on-site, not at the manufacturer's location. It is anticipated that ACP supervisory personnel will receive factory training and then develop an on-site training program to be utilized for on-site training of ACP maintenance personnel; additional on-site training provided by the manufacturer may be an option if deemed appropriate.
Section 8.4.3.a) 1)	Battery charging system inspections are anticipated to be monthly in accordance with Table 8-1, not weekly.
Section 8.4.3.a) 2)	The diesel-generator (D-G) system testing will not consist of full-load, weekly testing. A plant procedure for periodic testing of the D-G set will be developed in accordance with existing plant D-G testing practices based upon nearly 50 years operating experience and the D-G manufacturer's recommendations.

Section 8.5.2 Daily inspections of uninterruptible power supply (UPS) systems will not be required; inspections are anticipated to be monthly in accordance with Section 8.5.2.b.

Section 8.5.2.a) The listed UPS "weekly inspection" items are anticipated to be monthly and included in the routine inspections listed in Section 8.5.2.b).

Section 8.6.1 A battery system maintenance procedure will be developed in accordance with existing plant battery system practices based upon nearly 50 years operating experience and the battery system manufacturer's recommendations. It is anticipated that general battery system inspections will be performed monthly in accordance with Table 8-1.

Section 8.9 The Licensee does not commit to all of the standards listed in this section.

Sections 10.4 a.) thru c.) The UPS final factory testing steps will be based upon the capacity (size) of the system, the precise type of batteries, the system configuration, and the intended function of the installed system.

Section 10.9 The Licensee does not commit to all of the standards listed in this section.

For the reference to this standard see Sections 2.6.4 and 2.6.7 of the ISA Summary for the ACP.

- *IEEE* 484-2002, *IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications*

  The Licensee will satisfy the provisions of this standard.

  For the reference to this standard see Section 3.8.9 of the ISA Summary for the ACP.

- IEEE 603-1998, *Standard Criteria for Safety Systems for Nuclear Power Generating Stations*

  The Licensee commits to utilizing IEEE 603 Clauses 1 (Scope), 3 (Definitions) and 7 (Execute Features) and portions of Clauses 5 (Safety System Criteria), 6 (Sense and Command Features), and 8 (Power Source Requirements).

  The Licensee takes exception to the contents of IEEE 603 Clauses 2 (References), 4 (Safety System Design Basis), and Annexes A, B, and C. These clauses are not considered to be applicable or necessary due to their nuclear reactor content and redundancy with other IEEE standards and the Licensee's ISA. Annexes A, B, and C provide only "informative" details and references. In addition, the Licensee takes exception to portions of contents in IEEE 603 Clauses 5, 6, and 8 for the following reasons:

Sections 5 and 5.1	Single-failure criterion will be applied only where needed to provide the reliability of the IROFS credited in the ISA.
Sections 5.3 and 5.3.1	The Licensee commits to ASME NQA-1-2008 with addenda Part I, Requirement 11 and Part II, Subpart 2.7 as defined in Section 1.4.3 of this license application.
Section 5.4	Qualification - Use and qualification of equipment is specified in the Licensee's IEEE 323 commitment above.
Sections 5.6.1 and 5.6.2	The Licensee's goal is to design any safety system that might not survive all design basis events such that it is electrically failsafe (i.e., does not require electrical power to perform its intended safety function).
Section 5.15	Reliability analysis methods and calculations are as specified in the ACP ISA. The ACP condition notice system will be monitored and evaluated.
Section 6.2	Manual control requirements may not be applicable to all IROFS; the need will be evaluated during the final design phase.
Section 8.1	Safety systems that are failsafe upon loss of electrical power will not require redundant power sources.

For the reference to this standard see Sections 2.6.4 and 2.6.10 of the ISA Summary for the ACP.

- IEEE 1023-2004, *IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities*

  The Licensee will satisfy the provisions of this standard.

  For the reference to this standard see Section 2.6 of the ISA Summary for the ACP.

- IEEE 1050-1996, *Guide for Instrumentation and Control Equipment Grounding in Generating Stations*

  The Licensee commits to utilizing IEEE 1050 Clauses 1 (Overview), 3 (Definitions), 4 (Design), 5 (System Grounding), 6 (Shield Grounding), and 7 (Testing).

  The Licensee takes exception to the contents of IEEE 1050 Clause 2 and Annexes A and B. The Licensee does not commit to all of the standards listed in Clause 2. Annexes A and B provide only "informative" references.

  For the reference to this standard see Section 2.6.4 of the ISA Summary for the ACP.

1.4.9 Other Various Codes and Standards

- ASCE 7-2002, *Minimum Design Loads for Buildings and Other Structures*

  The Licensee will satisfy the provisions of this standard.

  For the reference to this standard, see Sections 1.3.3.1 and 1.3.3.3 of this License Application.

- ANSI/ISA 67.04.01-2018 *Setpoints for Nuclear Safety-Related Instrumentation*

  The IROFS related setpoints are determined utilizing methodologies in accordance with this standard. The Licensee commits to utilizing ISA 67.04.01 Clause 1 (Purpose), 2 (Scope), 3 (Definitions), 4 (Establishment of Setpoints), 5 (Documentation), and 6 Maintenance of Safety-Related Setpoints).

  The Licensee takes exceptions to the contents of ISA 67.04.01 Clauses 7 (References) and 8 (Informative References). The Licensee does not commit to all the standards listed in Clauses 7 and 8.

  For the reference to this standard see Section 2.6.10 of the ISA Summary for the ACP.

1.5 License Application Regulatory Guidance Documents

The following sub-sections lists the various regulatory guidance documents that have been referenced in this license application. The extent to which the Licensee satisfies each guidance document is identified individually in the sub-sections.

1.5.1 U.S. 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 RG, 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 DOE reservation Emergency Plan.

  For references to this RG, see Section 8.0 of this license application. This RG currently does not apply under the HALEU Demonstration Program.

- Regulatory Guide 3.71, Revision 3, *Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Core*

  This RG endorses ANSI/ANS-8 standards. The Licensee commits to ANSI/ANS-8.1-2014, ANSI/ANS-8.3-1997, ANSI/ANS-8.19-2014, and ANSI/ANS-8.20-1991 as described above.

  For the reference to this RG, see Section 5.5 of this license application and Section 3.10.6 of the ISA Summary for the ACP.

- Regulatory Guide 5.80, Revision 0, *Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material.*

  The Licensee satisfies the provisions of this RG.

  For the reference to this RG, see Section 3.3.4 of Security Program for the American Centrifuge Plant.

- 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 4.7.3 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, Revision 0, *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.

- NUREG-1065, *Acceptable Standard Format and Content for the Fundamental Nuclear Material Control Plan Required for Low Enriched Uranium Facilities*

  This NUREG was used for general reference purposes in structuring the FNMCP for the ACP. This NUREG currently does not apply under the HALEU Demonstration Program.

  For references to this NUREG, see Section 15.0 of the FNMCP for the ACP.

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

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

  For references to this NUREG, see Sections 3.1.2, 3.2, 3.3, 5.5, 6.4, 7.2.2, 7.6, 8.2, 9.2.3, and 9.4 of this license application.

- NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications,* Revision 2

  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 structure of the NUREG.

  For references to this NUREG, see Sections 1.0, 1.4, 3.2, 5.5, 6.4, 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-1748, *Environmental Review Guidance for Licensing Actions Associated with NMSS Programs*

  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 the Environmental Report for the ACP.

- NUREG-1757, *Consolidated NMSS Decommissioning Guidance, Volumes 1, 2, and 3, Final Report.*

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

  For the references to this NUREG, see Section 10.10.1 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 10 of the FNMCP for the ACP.

- 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 this NUREG be followed.

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

  For the reference to this NUREG, see Section 8.7 of the FNMCP for the ACP.

- NUREG/BR-0096, *Instruction and Guidance for Completing Physical Inventory Summary Reports, NRC Form 327*

  This NUREG provides line-by-line instructions for preparing NRC Form 327, Special Nuclear Material and Source Material Physical Inventory Summary Reports.

  The Licensee satisfies the provisions of this NUREG.

For the reference to this NUREG, see Section 12.4 of the FNMCP for the ACP.

- NUREG/CR-4604, *Statistical Methods for Nuclear Material Management*

  This NUREG contains techniques and formulas used to estimate random and systematic error variances associated with nuclear material measurement methods.

  For the reference to this NUREG, see Section 9.1.1 of the FNMCP for the ACP.

- NUREG/CR-5734, *Standard Format and Content for the Fundamental Nuclear Material Control Plan Required for Low Enriched Uranium Enrichment Facilities*

  This NUREG is used to establish the Detection Quantity for evaluation of nuclear material inventory differences.

  For the reference to this NUREG, see Section 9.4 of the FNMCP for the ACP.

- 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 3.1.2.3.2.2.5.1 of this license application and Section 3.3 of the ISA Summary for the ACP.

- 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 6.4 of this license application.

1.5.2 Other Various Guidance Documents

- 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 2.0 of the QAPD for the ACP.

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

  The data contained in Tables 2-1 and 2-2 of this document used to calculate dose conversion factors for radionuclides of concern. This data is also used to calculate the Derived Air Concentrations (DACs) listed in Table 4.7-4.

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

- IAEA Safeguards Technical Manual, Part F, Volume 3

  The method used to establish sample sizes for item monitoring activities was obtained from this manual.

  For the reference to this recommended practice, see Section 7.4 of the FNMCP for the ACP.

1.6 References

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

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

16. ORO-EP-120, *Seismic Design Criteria for the Gas Centrifuge Enrichment Plant – GCEP*, U.S. Department of Energy Oak Ridge Operations Office, Office of the Deputy Manager for Enrichment Expansion Projects, Oak Ridge, Tennessee, August 1980

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19. USEC-651, "The UF_6 Manual – Good Handling Practices for Uranium Hexafluoride," Revision 9, July 2006

20. ASTM C1052, *Standard Practice for Bulk Sampling of Liquid Uranium Hexafluoride*, 2014

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2.0 ORGANIZATION AND ADMINISTRATION

The Licensee is committed to conducting operations at the American Centrifuge Plant (ACP) 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 objectives, as well as others required for operation of the ACP, 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, ACP personnel.

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

The Licensee has provided the management structure to ensure that this policy is effectively implemented and is responsible for the safe operation of the ACP. Programs are established for the environmental, health, safety, safeguards, security, and quality areas and are provided with sufficient resources to support safe operation of the ACP. Contracted resources are utilized in a number of these programmatic areas to provide day-to-day functional support. Arrangements (i.e., through reverse work authorizations) are in place to provide the necessary support.

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

Managerial positions that have the principal responsibilities important to environmental, health, safety, safeguards, security, and quality for the ACP 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, operation, and future decommissioning. 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 exercises stop work authority, the Senior Vice President, Field Operations 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 the programs.

Section 2.3 describes the plans and management controls for pre-operational testing and initial start-up of the ACP.

2.1 Organizational Commitments, Relationships, Responsibilities, and Authorities

The American Centrifuge 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 depicts the American Centrifuge organization.

Various day-to-day functional support for carrying out the requirements of the environmental, safety, health, and safeguards programs, and security plans may be provided by contractors (i.e., through reverse work authorizations), along with administrative services required to support overall facility operations. American Centrifuge management maintain overall decision-making authority and responsibility for oversight of the major functional support areas that may be provided by contractors. Contractors may also provide the necessary utilities (e.g., electricity, cooling water, potable water, and sanitary sewage) to support operations.

Minimum qualifications, functions, and responsibilities for key staff positions are described below. The personnel responsible for managing the design, refurbishment/construction, manufacturing, operation, and future decommissioning of the plant have the substantive breadth and level of experience to successfully execute their responsibilities. These key staff 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. Alternates will meet the minimum qualification for the corresponding position.

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 meet 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 plant). These factors are evaluated on a case-by-case basis, documented, and approved by the appropriate Director or General Manager.

2.1.1 Senior Vice President, Field Operations

The Senior Vice President, Field Operations reports to the President and Chief Executive Officer and has overall responsibility for the safe operation and the deployment of American Centrifuge Project(s), including facility design; process equipment procurement; machine design; testing, and manufacturing; enrichment plant refurbishment/construction; testing of facilities; and turn-over to operations. The Senior Vice President provides strategic leadership and direction for the enrichment operations organization, including the functions of operations; maintenance; project support; engineering; system(s) testing; transportation; procurement; materials handling and storage; industrial, radiological, and nuclear safety; and future decommissioning. The individual also has overall responsibility for the development and implementation of conduct of operations for the ACP and associated plans, programs, and management measures as defined by the regulatory requirements. The Senior Vice President is responsible for the QA program and for determining the status, adequacy, and effectiveness of the Quality Assurance Program Description (QAPD).

The General Manager; Director, Quality Assurance; Director, Engineering, Procurement, and Construction (EPC); Director, Nuclear Safety; and Director, Engineering report to the Senior Vice President and manage the activities in their areas of responsibility.

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

2.1.2 General Manager

The General Manager reports to the Senior Vice President, Field Operations. The General Manager is responsible for the day-to-day safe operation of the plant, including direction of operation and maintenance of the ACP; overall responsibility for the Plant Safety Review Committee (PSRC), Nuclear Safety, and Radiological Protection program for keeping exposures and contamination below regulatory limits and as low as reasonably achievable; 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 ACP operations, 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 Regulatory Manager, Business Services Manager, Operations Manager, and Production Support Manager report directly to the General Manager and manage the activities in their area of responsibility. Additionally, the Piketon Quality Assurance Manager; Industrial Safety Manager; Director, EPC; Director, Nuclear Safety; Piketon Engineering Manager have matrixed responsibilities directly to the General Manager in support of safe operations at Piketon ACP facilities.

The General Manager 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.

2.1.2.1 Regulatory Manager

The Regulatory Manager reports to the General Manager and is responsible for regulatory oversight functions and commitment management. The Regulatory Manager, as delegated by the Senior Vice President and General Manager, maintains the day-to-day interface with NRC representatives on matters of regulatory compliance. This manager has responsibility for maintaining the change evaluation process and ensuring the change evaluation reporting requirements are met. The Regulatory Manager is also responsible for implementing the Corrective Action Program; ensuring incident investigations are performed and providing management with data to assure that corrective actions and commitments are properly addressed and managed to facilitate compliance with the implementing policies and procedures. The Regulatory Manager is also responsible for the Nuclear Materials Control and Accountability (NMC&A) program that is independent from operations.

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

2.1.2.1.1 Nuclear Materials Control and Accountability Manager

The NMC&A Manager reports to the Regulatory Manager and has programmatic responsibility for the NMC&A program, . ensuring regulatory requirements are met on a day-to-day basis. This manager is independent from production, plant operating cost, and production schedule concerns. This manager has direct access to the General Manager for resolution of concerns dealing with the NMC&A Program.

The NMC&A Manager has, as a minimum, a bachelor's degree in engineering or a technical field or equivalent technical experience, and experience in nuclear materials safeguards.

2.1.2.2 Business Services Manager

The Business Services Manager reports to the General Manager and has matrixed responsibilities for procurement; packaging, transportation, and materials management; finance; and information technology in support of the American Centrifuge Project(s).

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.

2.1.2.2.1 Procurement Manager

The Procurement Manager reports to the Director, Engineering, Procurement and Construction and is responsible for providing support services to the Business Services Manager for procurement and providing procurement material control services (including supplier qualification coordination, purchasing, contracting). This manager is also responsible for supply strategy and development of qualified long-lead-time and complex-system suppliers.

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.

2.1.2.2.2 Packaging, Transportation, and Materials Management Manager

The Packaging, Transportation, and Materials Management Manager reports to the Director, Engineering, Procurement and Construction and is responsible for providing support services to the Business Services Manager for packaging and transportation of classified matter and radioactive material.

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

2.1.2.3 Operations Manager

The Operations Manager reports to the General Manager and is responsible for fissile material operations, centrifuge operations, and shift operations. This manager is responsible for directing activities of the Cascade / Recycle and Assembly Operations Shift Supervisors in operation of the cascade, feed and withdrawal, and gas test, as well as the Maintenance Work Center Supervisor for maintenance and operations of the plant equipment, utilities processes, and facilities. This includes centrifuge assembly, drying, transportation, and installation in the cascade; safe operation of the uranium hexafluoride (UF_6) processes in accordance with approved procedures; proper receipt, storage, handling, and onsite transportation of UF_6; execution of the Integrated Systems and Test Plans (ISTPs), initial start-up, and operation of the centrifuges, equipment, and support systems. Other activities include select repair of centrifuges; maintenance; classified equipment control; accountable property inventory, segregation, and disposition; contractor support; integrated planning and scheduling; caretaker activities; materials management support; and future decommissioning and disposal activities, ensuring activities are performed in accordance with approved programs, processes, and procedures.

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.

2.1.2.3.1 Integrated Systems Test and Start-up Manager

The Integrated Systems Test/Start-up Manager reports to the Operations Manager and is responsible for assisting in the development of and execution of the ISTPs which demonstrate the proper operation of completed systems to ensure that the systems meet their intended design functions. This manager is also responsible for the acceptance of turnover from the EPC or from contractors/vendors to the Licensee; initial acceptance testing; and initial start-up of equipment and support systems.

The Integrated Systems Test/Start-up 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.

2.1.2.3.2 Process Area Managers [commercial operations only]

Process Area Managers report to the Operations Manager and are responsible for directing activities of the Cascade, Recycle and Assembly, and Balance of Plant (BOP) Operations Shift Supervisors in 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 UF_6 processes, proper receipt, storage, handling, and on-site transportation of UF_6; machine installation and pump down; integrated system testing; provide oversight in the areas of BOP Operations and Facility Surveillances; and future Construction Work In Process (CWIP) Waste Disposition; and Classified Equipment Control and Centrifuge Disposition.

These Process Area Managers are responsible for the plant utilities operations, process and facility surveillances; CWIP and waste disposition, classified equipment control, centrifuge storage, transport, disassembly, and disposition; UF_6 cylinder storage, handling, transportation, and disposition; shift operations; accountable property assessment, inventory, and segregation; and caretaker operations. The Process Area Managers are also responsible for directing the activities of the Cascade / Recycle and Assembly Operations Shift Supervisors to accomplish these objectives and includes activities such as ensuring the safe operation of the plant utilities operations and the future disassembly, decommissioning, and disposition of materials.

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.

2.1.2.3.3 Cascade / Recycle and Assembly Operations Shift Supervisors

Cascade / Recycle and Assembly Operations Shift Supervisors report to the Operations Manager and are responsible for directing the operation of systems within the facilities necessary to support facility operation within approved programs, processes, and procedures. The Cascade / Recycle and Assembly Operations Shift Supervisors authorize the restart of equipment that has been shut down in a routine fashion when the prerequisites and limitations of the associated operating procedure are met. The Cascade / Recycle and Assembly Operations Shift Supervisors are responsible for providing operational support of centrifuge assembly, transport, installation, pump down, integrated system testing, start-up, operation, disassembly, and select repair. The Cascade / Recycle and Assembly Operations Shift Supervisors also direct the operation of systems with the facilities, necessary to support the operation and future decommissioning activities.

As the senior manager on shift (one per shift), the Cascade / Recycle and Assembly Operations Shift Supervisor represents the General Manager and has the authority and responsibility to make decisions, as necessary, to ensure safe operations. These supervisors are responsible for accumulation and dissemination of information regarding American Centrifuge activities to the Incident Commander during emergencies.

Cascade / Recycle and Assembly 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.

2.1.2.3.4 Senior Shift Supervisors [commercial operations only]

Senior Shift Supervisors report to the Operations Manager. As the senior manager on shift (one per shift), the Senior Shift Supervisor represents the General Manager and has the authority and responsibility to make decisions, as necessary, to ensure safe operations. The Senior Shift Supervisors are responsible for accumulation and dissemination of information regarding American Centrifuge activities to the Incident Commander during emergencies and making notification of events to regulatory agencies. The Senior Shift Supervisors are also responsible for directing the operation of systems within the facilities necessary to support enrichment operation and future disassembly, decommissioning, and disposal activities and caretaker operations. The Senior Shift Supervisors authorize the restart of equipment that has been shut down in a routine fashion when the prerequisites and limitations of the associated operating procedure are met.

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

2.1.2.3.5 Maintenance Work Center Supervisor

Maintenance Work Center Supervisor reports to the Operations Manager. The Maintenance Work Center Supervisor is responsible for directing activities of the BOP Operations Shift Supervisors and of the Maintenance Shift Supervisors in the performance of preventive, predictive, and corrective maintenance and to provide support on facilities and equipment, within approved programs, processes, procedures, and personnel training limitations. 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.

The Maintenance Work Center Supervisor is also responsible for integrated planning, scheduling, and materials management. This includes maintenance of logs and records; managing daily work control activities; maintenance of an integrated work schedule to initiate, screen, evaluate, and prioritize maintenance work; coordinating shop maintenance activities; and coordinating development of work control guidelines.

Maintenance Work Center Supervisor has, 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. Maintenance Work Center Supervisors must have one year of supervisory experience or completion of a supervisory training course.

2.1.2.3.5.1 Balance of Plant Operations Shift Supervisors

BOP Operations Shift Supervisors report to the Maintenance Work Center Supervisor and are responsible for directing the activities for plant utilities processes and facilities within approved programs, processes, and procedures.

BOP 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. BOP Operations Shift Supervisors must have one year of supervisory experience or completion of a supervisory training course.

2.1.2.4 Production Support Manager

The Production Support Manager reports to the General Manager. This manager is responsible for fire safety; emergency management; radiation protection (RP), which includes chemical process safety, health physics, industrial hygiene, and environmental/waste management; security; and training and procedures, which includes records management and document control. During commercial operations, this manager will also be responsible for the Customer Order Management program.

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

2.1.2.4.1 Fire Safety / Emergency Management Manager

The Fire Safety/Emergency Management Manager reports to the Production Support 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.

2.1.2.4.2 Radiation Protection Manager / Supervisor

The Radiation Protection Manager (RPM)/Supervisor reports to the Production Support Manager. The RPM/Supervisor 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 by personnel who do not adhere to radiological protection requirements. The RPM/Supervisor also has oversight of radiological protection procedures in order to maintain the integrity of the RP Program. The RPM/Supervisor has direct access to the General Manager and the Senior Vice President for RP matters.

This position also has programmatic responsibilities for chemical process safety, health physics, industrial hygiene, and environmental/waste management activities.

The RPM/Supervisor 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 of prior Radiation Protection Manager/Supervisor experience at a nuclear facility.

2.1.2.4.3 Security Manager

The Security Manager reports to the Production Support Manager. This manager is responsible for the strategic direction of the site security operations and programs for safeguards and security services. The Security Manager has direct access to the General Manager and Senior Vice President for security matters.

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

2.1.2.4.4 Training and Procedures Manager

The Training and Procedures Manager reports to the Production Support 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 the programmatic oversight of the Records Management and Document Control (RMDC) programs.

The Training and Procedures 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.

2.1.2.4.4.1 Records Management and Document Control Manager

The RMDC Manager reports to the Training and Procedures Manager. This manager is responsible for the RMDC programs.

The RMDC 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 Director, Quality Assurance

The Director, QA reports to the Senior Vice President. This Director is a member of the senior management team of the American Centrifuge Project and has been designated the responsibility for ensuring that the project achieves its quality targets and meets its regulatory driven quality commitments in a safe manner. This Director is responsible for QA for the operations, including future decommissioning as applicable, at the Piketon, Ohio and Oak Ridge, Tennessee facilities; for vendors and suppliers; and for construction and manufacturing activities, both for internal and external customers.

This Director advises and provides guidance to the Senior Vice President on matters of safety and QA. The Piketon QA Manager and Industrial Safety Manager report to the Director, QA and are independent from production, plant operating cost, and production schedule concerns to ensure appropriate independent oversight of project activities.

The Director, QA 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.

2.1.3.1 Piketon Quality Assurance Manager

The Piketon QA Manager reports to and receives technical direction for QA matters from the Director, QA and is matrixed directly to the General Manager. The Piketon QA Manager has the responsibility to exercise oversight of design, procurement, refurbishment/construction, manufacturing, testing, start-up, plant operations, maintenance, and future decommissioning to ensure that the health and safety of the public and workers are adequately protected; to ensure compliance with safety, safeguards, and quality requirements; and to ensure implementation of the QAPD, policies, and procedures. The Piketon QA Manager provides independent assessment and audit of ACP activities.

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

The Piketon 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.

2.1.4 Director, Engineering, Procurement, and Construction

The Director, EPC reports to the Senior Vice President and is matrixed directly to the General Manager. During the refurbishment/construction of the ACP, this director is responsible for providing technical administration and direction to the engineering, procurement, and construction contractor(s); and providing the primary interface with the refurbishment/construction contractor(s).

The Director, EPC 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 Director, Nuclear Safety

The Director, Nuclear Safety reports to the Senior Vice President and is matrixed directly to the General Manager. This director is responsible for developing and implementing the nuclear safety program, including technical oversight of nuclear safety, including nuclear criticality safety (NCS) and maintenance of the Integrated Safety Analysis (ISA), safety analysis training, review of procedures involving fissile material operations, and assessments of program implementation. This director is also responsible for direct management of the NCS functions and administration of the NCS program on a day-to-day basis. These activities may include conducting assessments of nuclear safety 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.

The Director, Nuclear Safety has, as a minimum, a bachelor's degree in engineering, mathematics, or related science or equivalent technical experience, and six years nuclear experience.

2.1.6 Director, Engineering

The Director, Engineering reports to the Senior Vice President and has the overall responsibility for successful deployment of the centrifuge technology in an operational plant environment. This director is the overall design authority for Piketon operations. This director provides strategic leadership and direction to the engineering organization and manages the utilization of engineering resources across the enterprise to support field operations. This director has design authority for the American Centrifuge operations. Design authority is then delegated to the Piketon Engineering Manager to provide day-to-day engineering support.

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

2.1.6.1 Piketon Engineering Manager

The Piketon Engineering Manager reports to the Director, Engineering. This manager is the delegated design authority for Piketon operations and is matrixed directly to the General Manager. This manager is responsible for Piketon engineering activities in support of operations and future decommissioning, which includes maintaining the configuration management program; systems and design engineering; review of design and modifications of items relied on for safety (IROFS); and supporting procurement services. This manager is also responsible for the development of the ISTPs.

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

2.1.6.1.1 Configuration Management Manager

The Configuration Management Manager reports to the Piketon 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 plant are maintained.

The Configuration Management 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.

2.1.6.1.2 Piketon System Engineering Manager [commercial operations only]

The Piketon System Engineering Manager reports to the Piketon Engineering Manager. This manager has responsibility for the system engineering activities in support of plant operations.

The Piketon 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.

2.1.6.1.3 Piketon Design Engineering Manager [commercial operations only]

The Piketon Design Engineering Manager reports to the Piketon Engineering Manager. This manager has responsibility for the design engineering activities in support of plant operations, which includes providing engineering support and review of the design and modifications of IROFS.

The Piketon Design 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.

2.1.7 Plant Shift Superintendent (Contractor)

The Plant Shift Superintendent (PSS) reports to the U.S. Department of Energy (DOE) reservation contractor management and provides support through approved reverse-work authorizations with the DOE. 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 a centralized point for incident identification, screening, and reporting. The PSS's responsibilities are consistent with those exercised at the gaseous diffusion plant 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 gaseous diffusion plant, or a high school diploma plus 12 years experience at a gaseous diffusion plant.

2.1.8 Shift Crew Composition [only during operational phases with licensed material]

The minimum operating shift crew consists of an Operations Shift Supervisor, a Radiation Protection/Industrial Hygiene technician, and one operations technician per process building. Other personnel, such as NCS, will be available on an as needed basis.

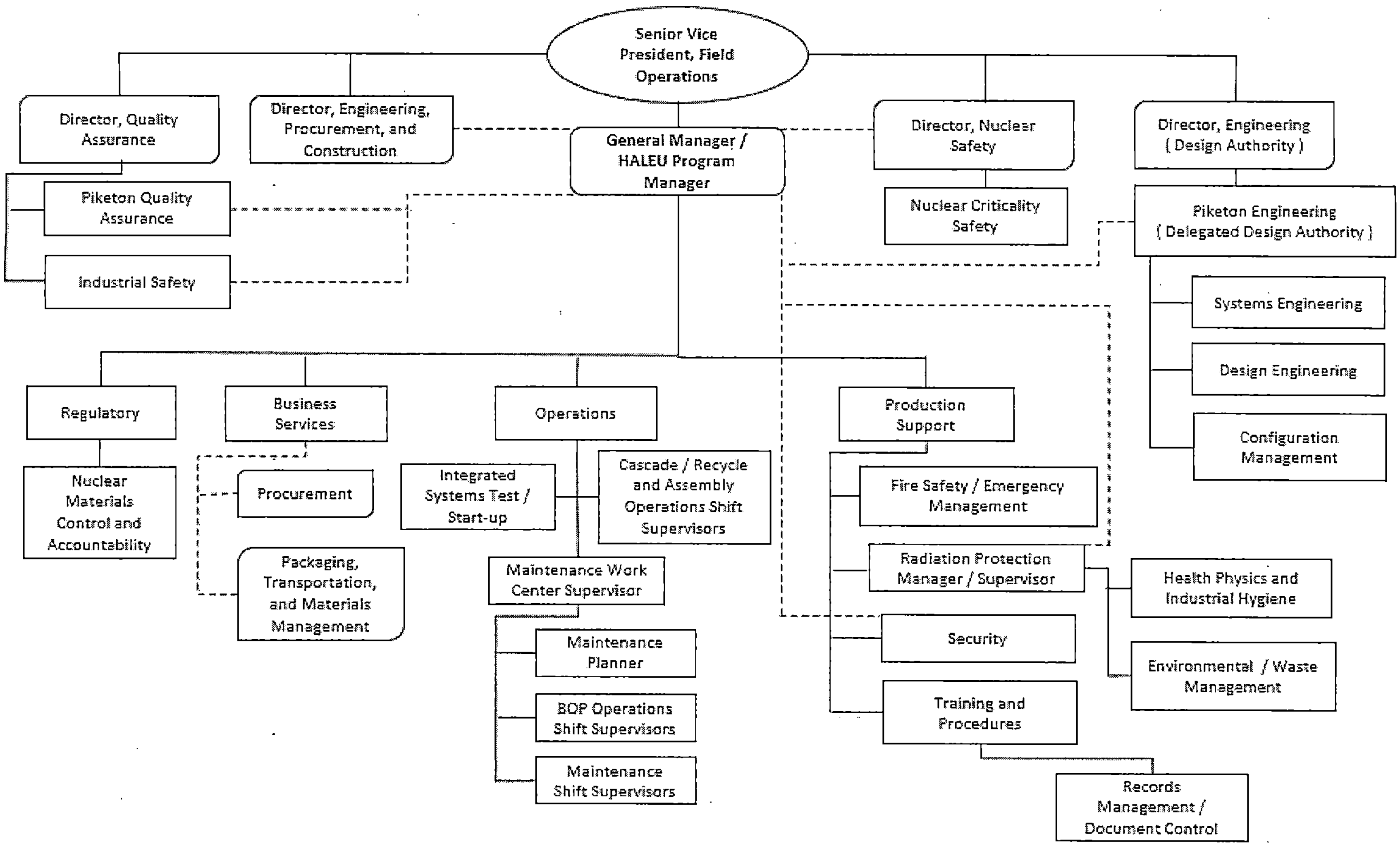

Figure 2.1-1
American Centrifuge Organization Chart

2.2 Management Controls

The Licensee has established management measures with associated policies, administrative procedures, and management controls to ensure the ACP equipment, facilities and procedures; the staff (including training and qualifications); and the programs provide for the protection of the health and safety of workers and the public, protection of the environment, and for the common defense and security. Management controls have been established to maintain configuration management of the plant. These controls are described in Section 11.1 of this license application. Organizations with environmental, health, nuclear 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 plant are clearly defined.

The management controls established for the ACP include policies, management systems, and administrative procedures that are communicated to plant 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 and security 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 5.0 and 6.0 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 plant programs are identified in the QAPD, Fundamental Nuclear Material Control Plan, and Security Plans.

The commitment tracking and Corrective Action Programs are integrated to prioritize ACP actions consistent with their safety and safeguards significance. Any person working in the plant may report potentially unsafe conditions or activities by submitting a condition notification. 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 ACP 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 PSRC; and

- 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.

Letters of agreement exist with off-site emergency resources (i.e., fire, police, ambulance/rescue units, and medical services).

2.2.1 Plant Safety Review Committee

The PSRC performs multi-discipline reviews of day-to-day and proposed activities to ensure that these activities are and/or will be conducted in a safe manner. The PSRC advises the General Manager 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 plant procedure. Auditing and oversight of PSRC activities is the responsibility of the Piketon QA Manager.

Subcommittees may be established by the PSRC chairperson to provide assistance in conducting reviews and assessments as described in the PSRC procedure. The PSRC chairperson approves the subcommittee procedures, membership, and member qualifications. The PSRC 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 centrifuges, equipment, and support systems. These plans cover the transition from the refurbishment/construction phase to the operations phase.

The Integrated Systems Test/Start-up Manager is responsible for development and implementation of plans to provide for the turnover and testing of equipment and systems from contractors/vendors to the Licensee.

The Piketon Engineering Manager is responsible for the development of ISTPs with the assistance of the Integrated Systems Test/Start-up Manager. The Integrated Systems Test/Start-up Manager is responsible for the execution of the ISTPs. The ISTPs demonstrate the proper operation of completed systems to ensure the systems meet their intended design functions. The Integrated Systems Test/Start-up Manager is also responsible for the acceptance of turnover from the EPC, initial acceptance testing, and initial start-up of equipment and support systems. The Operations Manager is responsible for the acceptance of turnover, initial acceptance testing, initial start-up, and operation of the centrifuges. 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 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; and
- Can be operated in a dependable manner so as to perform their intended functions.

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

The refurbishment/construction contractor(s) 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 procedures, design documents, and installation specifications. As systems or portions of systems are turned over to the Licensee, initial acceptance testing is performed in accordance with established schedules. The Integrated Systems Test/Start-up Manager is responsible for coordination of initial turnover and initial acceptance testing.

Integrated systems testing, as a minimum, includes system or component tests required by the pertinent design codes or QAPD that were not performed by the refurbishment/construction contractor(s) prior to initial turnover to 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 in a safe and controlled manner. The Integrated Systems Test/Start-up Manager is responsible for the execution of the ISTPs for the ACP. 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 during the operations phase and are the responsibility of the Operations Manager. Testing and turnover in conjunction with modifications identified by the Operations Manager following transition to the operations phase are the responsibility of the Piketon Engineering Manager.

2.4 References

None

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 be conducted and maintained. An ISA Summary is required by 10 CFR 70.65(b). Accordingly, the Licensee has conducted an ISA of adequate complexity to support preparation of an ISA Summary for the American Centrifuge Plant (ACP), including an Addendum to the ISA Summary that provides information specific to the HALEU Demonstration. The ISA is a compilation of the design and analysis documentation utilized to: 1) identify the potential accident sequences that could occur, 2) designate items relied on for safety (IROFS) to either prevent such accidents or mitigate their consequences to an acceptable level, and 3) identify the management measures to provide reasonable assurance of the availability and reliability of IROFS.

The ISA Summary is a synopsis of the ISA and contains the information required by 10 CFR 70.65(b). The ISA Summary is updated to reflect changes to the ISA. Neither the ISA nor the ISA Summary is incorporated as part of this license. The ISA documentation is available to the U.S. Nuclear Regulatory Commission (NRC) by request at the ACP through the Regulatory Manager. The ISA Summary (Reference 1), and its Addendum for the HALEU Demonstration (Reference 21), are maintained as a separate documents from the license application and are submitted separate from this license application. In addition to providing a synopsis of the results of the ISA, the ISA Summary and its Addendum describe the methods and criteria utilized in the safety analysis and describes the qualifications of the team performing the ISA.

In the context of this chapter, the general use of the term ISA Summary is intended to include the ISA Summary for the commercial ACP deployment (Reference 1) as well as the Addendum (Reference 21) that is uniquely associated with the HALEU Demonstration. Information that is applicable only to the commercial ACP operation will be noted as "non-HALEU" or "commercial ACP"; whereas, aspects that are unique to the HALEU Demonstration will be noted as "HALEU". References to specific tables or sections in the ISA Summary are intended to refer to those entries in Reference 1.

3.1 Safety Program and Integrated Safety Analysis Commitments

3.1.1 Process Safety Information

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 ACP to assure that chemicals not related to the licensed material are evaluated as necessary.

The ISA considers process safety throughout the analysis development. Process safety is considered when identifying the credible accident scenarios, developing the IROFS, and establishing the management measures to ensure the health and safety of the workforce and public. The ISA and ISA Summary are maintained and updated by written procedures using qualified personnel to ensure that process safety information is accurately reflected in accordance with 10 CFR 70.72. The license should be conditioned as follows: Upon completion of the design and updating of the appropriate documentation involving process safety information, the Licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in the American Centrifuge Plant in order to conduct its inspections involving process safety information that are required by 10 CFR 70.32(k). It is acknowledged that the ACP is a modular process that may be deployed in phases, such that notice to introduce UF_6 may be issued for approval to begin operations in a portion of the ACP (e.g. notice may be issued for deployment of the HALEU demonstration, independent of the full deployment of all other modules of the complete ACP.)

3.1.2 Integrated Safety Analysis

An ISA of the design and operation of the ACP 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 the design documentation and programmatic information reviewed and utilized during the course of the ISA effort. This information is available on site for NRC review.

The ISA documentation is sufficiently detailed to identify the following:

- 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;
- Credible accident sequences;
- Consequences and likelihood of each accident sequence;
- IROFS including the assumptions and conditions under which they support compliance with the performance requirements of 10 CFR 70.61; and
- Management measures.

Should the addition of new processes or other changes to the ACP be necessary, evaluations of appropriate complexity for each process will be performed in accordance with 10 CFR 70.72, using established ISA methods to ensure the processes can be carried out in a manner such that compliance with the performance requirements of 10 CFR 70.61 are maintained. The ISA methods utilized for the ACP are described in Section 3.1.2.1 of this license application.

The Licensee maintains the ISA and 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. ACP procedures specify the criteria for changing the ISA Summary. Changes to the ACP are evaluated against the ISA and ISA Summary 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 changes made to the ACP 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 is evaluated by means of the ISA methods. The Licensee evaluates proposed changes to the ACP or its operations by means of the ISA methods and designates new or additional IROFS, along with appropriate management measures, as necessary. The Licensee will periodically review IROFS per the requirements of 10 CFR 70.62(a)(3) to ensure their availability and reliability for use, and consistency with the ISA. As the final design is developed for the ACP, the management system and design approach will require that the final designs be reviewed against the ISA to ensure the ISA accurately reflects the ACP design and operations, identifies the credible accident sequences and appropriate assumptions, and credits the IROFS necessary to meet the performance requirements of 10 CFR 70.61. The license should be conditioned as follows: Upon completion of the design and updating of the ISA and ISA Summary, the Licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in the ACP (or into an operational module of the ACP, such as the HALEU Demonstration) in order to conduct its inspections involving the ISA and ISA Summary that are required by 10 CFR 70.32(k).

The Licensee also evaluates the adequacy of existing IROFS and associated management measures and makes any required changes to the ACP 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 IROFS and associated management measures are required, or if new IROFS or management measures are required. For any changes that require prior NRC approval under 10 CFR 70.72, the Licensee will submit an amendment request in accordance with 10 CFR 70.34 and 70.65.

The Director, Nuclear Safety 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 and coordinating information requests from the NRC.

Suitably qualified personnel update and maintain the ISA and ISA Summary. The ISA Team consists of at least one team leader who is formally trained and knowledgeable in the ACP's ISA methods and individuals with specific, detailed experience in 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 and ISA Summary. Written procedures are used to implement the ISA process and are maintained onsite. For any revisions to the ISA Summary, personnel having qualifications similar to those of ISA Team members who conducted the original ISA are used.

3.1.2.1 Integrated Safety Analysis Methodology

The ISA analyzes the hazards associated with ACP operation, its associated direct support equipment and support systems, and the buildings and facilities where it is located. This analysis does not address hazards associated with sabotage, chemical hazards that do not result from the processing of licensed nuclear material or have the potential for adversely affecting radiological safety, or Standard Industrial Hazards as presented in Section 3.1.2.3.1.3.2 of this chapter.

3.1.2.2 Selection of Evaluation Method

The guidelines presented in Appendix A of NUREG-1513 (Reference 2) serve as a basis for selecting the Hazard Evaluation Method, using the methodology in the flowchart, Figure A.1 of NUREG-1513. The method is selected using accepted evaluation techniques, experience, and judgment. The evaluation method for commercial ACP operation was selected by answering the questions at each decision branch which led to a selection of the Preliminary Hazard Analysis (PHA) method or the What-If/Checklist (WI/CL) method of analysis.

As a result, the ISA Team selected a hybrid method that incorporated elements of both the WI/CL and PHA methods. The WI/CL method combines the broad spectrum of accidents that can be postulated by a brainstorming team of experts with the detailed and comprehensive structure provided by a systematic Hazard Identification and Event Category checklist. Additionally, the use of a tabular accident recording form borrowed from the PHA technique provides for the effective listing and presentation of accidents along with their causes, hazard category, risk assessment and potential preventive and mitigative controls. Per Section 3.1.2, evaluations of appropriate complexity are completed for each new process or other changes to the ACP in accordance with 10 CFR 70.72.

3.1.2.3 Description of Selected Integrated Safety Analysis Method

The selected Hazard Analysis (HA) method for the ISA involves a combination of the PHA and WI/CL methods, as discussed above, which incorporates an unmitigated and mitigated approach. The method and approach has the advantage of providing a comprehensive and systematic process for addressing baseline facility and process hazards and credible accidents associated with those hazards, while the process and facility are still in the conceptual or preliminary design stages, thus helping to identify early in the design process those controls that are necessary to protect the public and workers.

The HA provides a systematic analysis of potential process-related, and external hazards including natural phenomena, that can affect the public and facility workers. The analysis considers the potential for both equipment failure and human error. In performing the HA, the ISA Team provides a thorough, predominantly qualitative evaluation of the spectrum of risks to the public, the workers, and the environment due to accidents involving the identified hazards. NUREG-1513 and NUREG-1520 (References 2 and 3) state that the hazard analyses comprehensively identify credible accidents and their causes, and estimate the frequency and consequences. Estimates of consequences and frequencies are performed in the hazard analysis such that attention is focused on those scenarios that have risk to the public, workers and the environment that exceeds the 10 CFR 70.61 performance requirements.

The Hazard Analysis for the ISA is developed using two primary activities:

- Hazard Identification
- Hazard Evaluation

3.1.2.3.1 Hazard Identification

Hazard Identification is a comprehensive and systematic process by which all known hazards (hazardous materials and energy) associated with the facility and process are identified, recorded, and screened by the ISA Team. In the HA, screening is performed to eliminate material/energy types and quantities that are considered "common hazards".

The Hazard Identification is divided into three steps:

- Sectioning of the facility;
- Facility information gathering and walkdowns; and
- Screening for Standard Industrial Hazards.

3.1.2.3.1.1 Sectioning the American Centrifuge Plant

Partitioning of the facility into "sections" facilitates hazard identification and evaluation. These sections may be based on specific operations, individual or grouped facility systems, specific function(s), types of material being handled, and/or physical boundaries inside the facility. In this process, interactions between the facilities are considered in the analysis to assure that the full range of events is evaluated.

The hazard identification and evaluation process applied to the commercial ACP operation included partitioning of the facility into the following sections:

- Cylinder Storage Areas (CY)
- Feed Area of Feed and Withdrawal Building (FB)
- Interconnecting Process Piping (FP)
- Process Buildings (PB) includes Process Support Building
- Withdrawal Area of Feed and Withdrawal Building (WS)
- Recycle/Assembly Building/Centrifuge Training and Testing Facility/Interplant Transfer Corridor (RA)
- Customer Services Building (BT)
- Transportation Activity (TA)
- Feed and Product Shipping and Receiving Building (SR)
- Criticality Events (CE)

The hazard identification and evaluation tables presented in the ISA Summary Appendices use the ACP section acronym identifiers as noted above. The hazard identification and evaluation process considered the applicable ACP activities including startup, normal operation, shutdown, and maintenance activities, as well as potential concurrent construction activities. The hazard identification and evaluation process performed for HALEU Demonstration is documented in *Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration* (Reference 21).

3.1.2.3.1.2 Information Gathering and Walkdowns

Facility information gathering is the key element in the process of identifying hazardous materials and energy sources that are currently known or which may be associated with each facility section, particularly at the conceptual design stage of a project. This information gathering process includes "paper walkdowns," which consist of a team review of current design documentation, system drawings, functional performance requirements, procedures, etc., in the context of Hazard Identification. In addition, the process uses direct interactions with the designers and/or system engineering personnel responsible for the specific sections of the facility. Also, if the design involves a modification to an existing facility, it is generally helpful to perform a physical walkdown of the facility as well to aid in the identification of potential hazards. The ISA Team uses a comprehensive hazards checklist that provides a structured method for conducting hazard identification. A sampling of items included on the checklist is shown in Table A-1 in Appendix A of the ISA Summary.

Using the results of the information gathering process, including paper and physical walkdowns and designer or operator interviews, the ISA Team creates a comprehensive list of all expected hazards, including radiological hazards and chemical hazards. The completed Hazard Identification Tables, as provided in Appendix B of the ISA Summary, are used to document the results of the Hazard Identification process and are developed for each facility section.

The ACP ISA Team hazards analysis and evaluation process used design and process information available from the various feasibility studies performed for the ACP as well as existing design, process, and safety analysis documentation applicable to the Gaseous Diffusion Plant (GDP) for those facilities, systems or processes similar to the ACP. Additionally, the ACP ISA Team performed physical facility walkdowns and observation of the current GDP facilities and operations including those used for feed, sampling and withdrawal processes and cylinder storage. Existing facilities proposed for use with the ACP were also walked down including the process buildings used for the GDP and facilities proposed for use as feed, blending, and transfer operations.

3.1.2.3.1.3 Screening of Chemical and Standard Industrial Hazards

The third step in the Hazard Identification process is the screening of chemical hazards and standard industrial hazards.

3.1.2.3.1.3.1 Chemical Hazards

At NRC-licensed fuel cycle facilities, the unacceptable consequences of concern (within NRC's regulatory authority) include those that result in the exposure of workers or members of the public to excessive levels of radiation or acute chemical exposure from licensed material or hazardous chemicals produced from licensed material. The mechanism for a radiological exposure could be a release of radioactive material, or an inadvertent nuclear chain reaction involving special nuclear material (criticality). The release of hazardous chemicals must also be addressed to the extent that such hazardous releases have the potential to cause, or reduce protection from, a radiation exposure accident. OSHA and EPA are responsible for regulating other aspects of chemical safety at the facility.

The consideration of radiological, including fissile, and chemical hazards includes radioactive materials, fissile materials, and chemical inventory, in all areas where such material is normally present or credibly could be present.

The ACP ISA Team examines each identified hazard for each section based on material/energy types and quantities and considers its potential contribution as an initiator for events involving release of radiological material, hazardous energy, or hazardous chemicals.

3.1.2.3.1.3.2 Standard Industrial Hazards

Standard Industrial Hazards are defined as hazards that are routinely encountered and accepted in general industry and construction, and for which national consensus codes and/or standards (e.g., OSHA or transportation safety) exist to guide safe design, operation or handling, without the need for special analysis for safe design and/or operational parameters. Typical examples would be slips, trips, and falls; routine industrial or construction noise; lifting equipment; welding equipment; and normal office hazards. They would also include substances and hazards that would be expected to be found for personal, family, or household use.

The following characteristics are used to classify hazards as standard industrial hazards:

- The hazard is controlled by OSHA regulations or national consensus standards (e.g., American Society of Mechanical Engineers, American National Standards Institute, National Fire Protection Association, Institute of Electrical and Electronic Engineers, National Electric Code), where these standards are adequate to define special safety requirements, unless in quantities or situations that initiate events with serious impact to the public or workers.

- Hazards such as noise, electricity, flammable materials, welding operations, small quantities of chemicals that would likely be found in homes or general retail outlets, and hazardous materials transported on the open road in DOT specified containers are considered to be common hazards encountered in everyday life.

Examples of common hazards/standard industrial hazards include:

- Specific materials (e.g., lead and asbestos) that have their own control program;
- Thermal energy sources (potential for burns);
- Electrical shock hazards;
- Gas cylinders transported and stored in DOT configuration;
- Personnel pinches, trips, falls, slips, etc.;
- Confined space hazards; and
- Hazards typically found in office areas.

3.1.2.3.2 Hazard Evaluation

The Hazard Evaluation (HE) constitutes the primary focal point of the HA. Hazards are characterized in the context of actual or anticipated facility operations and processes by considering feasible events, estimating event frequency, and estimating consequences of the event. The purpose of the HE is to ensure a comprehensive assessment of facility hazards and to focus attention on those events that pose the greatest risk to the public and on-site workers. The

HE described herein applies to facility hazards other than criticality; HE for criticality events is described in Section 3.1.2.3.2.7 for the commercial ACP (non-HALEU) and Section 3.1.2.3.2.8 for HALEU Demonstration. The scope of the HE includes:

- Identified aspects of facility process and operation.
- Natural phenomena (e.g., earthquakes, tornadoes, straight winds), other external events (e.g., aircraft and vehicular impact), facility events external to the process (e.g., fires, explosions), and process deviations, including failures of IROFS.
- Consideration of the entire spectrum of possible events for a given hazard in terms of both frequency and consequence levels.
- Hazards addressed by other programs and regulations (e.g., PSM, OSHA, *Resource Conservation and Recovery Act,* DOE, EPA) if loss of control of the hazard could result in a release of radiological material/hazardous chemicals or a nuclear criticality.

The scope of the HE does not include:

- Willful acts, such as sabotage.
- Hazardous events that meet the screening criteria given in Section 3.1.2.3.1.3.2 of this chapter.
- Events that would be associated with chemicals screened as described in Section 3.1.2.3.1.3.1 of this chapter.
- Events necessitating a change, either deliberate or inadvertent, to the design of the facility or process.

The HE process is divided into three steps:

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

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

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

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

3.1.2.3.2.1 Initial Conditions

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

In general, ICs represent assumptions made in the consequences or probability analyses, or specific passive and active design features credited in the probability analyses. Three examples are: 1) the header isolation features which serve to limit the material at risk as assumed in the consequence analyses (commercial plant only), 2) the combustible materials control program serves to limit the presence of material that could fuel facility fires, and 3) the structural seismic specifications serve to establish minimum structural requirements to reduce the frequency of certain events.

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

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

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

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

3.1.2.3.2.2 Unmitigated Hazard Evaluation

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

- Event Number and Category;
- Event Description (including location, release mechanism, material at risk, initial conditions specific to the event, and hazard source);
- Cause(s);
- Unprevented Event Frequency Level;
- Unmitigated Consequence Level (categorized as Low, Intermediate or High); and
- Unprevented/Unmitigated Risk Bin (categorized as A or B).

For an unmitigated analysis, estimated values are provided in the columns pertaining to Unprevented Event Frequency and Unmitigated Consequences. Additionally, any preventive and mitigative controls that may be available within the facility are listed in their respective HE Table columns as provided in Appendix C of the ISA Summary. However, no credit is taken for the available controls during the unmitigated hazard analysis (unless the control is listed as an Initial Condition).

3.1.2.3.2.2.1 Event Number and Category

In the HE Tables, events are identified by a unique sequential reference. The first two letters typically represent the facility section (~~i.e.,~~e.g., "PB" for ACP Process Building) as indicated in Section 3.1.2.3.1.1 above, the first number represents the event category as described below, and the second number (following the hyphen) represents the event sequential number.

Events are categorized according to the nature of the postulated release mechanism. Table A-3 in Appendix A of the ISA Summary provides some additional information regarding event categories and associated hazardous material and energy sources. The categories are as follows:

- Fire (Category 1)
- Explosion (Category 2)
- Loss of Containment/Confinement (Category 3)
- Direct Radiological/Chemical Exposure (Category 4)
- Nuclear Criticality (Category 5)
- External Hazards (Category 6)
- Natural Phenomena (Category 7)

3.1.2.3.2.2.2 Event Description

A brief description of a postulated event is given in this column of the HE Tables. The event description defines the nature of the event and includes the event type, location, release mechanism, Material-at-Risk (MAR), initial conditions (if applicable), and hazard source. Using the results of the Hazard Identification process as a basis, the ISA Team develops event scenarios for each facility system or area where a potential exists for a release of hazardous energy and/or material. The scenarios cover a broad spectrum of credible events for a given hazard; from low consequence events, for which procedures or equipment may be credited in providing adequate protection, to credible high consequence events. Events typically progress to and result in a release of hazardous material.

3.1.2.3.2.2.3 Cause

The event cause specifically states the failure, error, operational, and/or environmental condition that initiates the progression of occurrences that leads to the event. The cause(s) need to be clearly identified in order to support event frequency estimates. The cause(s) listed typically identify the major contributors and do not necessarily provide an exhaustive list of every possible cause. The Hazard Identification Tables (Appendix B of the ISA Summary) are used as a guide in developing specific causes for events. When multiple causes are apparent, they are separately numbered in the HE Table Cause column for the event.

3.1.2.3.2.2.4 Unprevented Frequency Level

3.1.2.3.2.2.4.1 Internal and External Initiated Events

Unprevented (sometimes termed "Unmitigated") frequency level evaluation is a predominantly qualitative (or semi-quantitative) process that involves assigning a frequency level to each event (event is defined as the progression of occurrences necessary to release hazardous material/energy, i.e., from initiator, through to the point of release) in the HE Tables. The term "unprevented" is used to designate an event frequency derived during the unmitigated HE before preventive features are credited to reduce the event frequency. Frequency levels with numerical descriptions, which are based on NUREG-1520, Section 3.4.3.2 (9) Quantitative Definitions of Likelihood (Reference 3) are summarized in Table A-4, Frequency Evaluation Levels in Appendix A of the ISA Summary. Specifically, a "Highly Unlikely" event is defined as an event with a frequency less than 10^{-5} occurrences per year, while an "Unlikely" event is defined as an event with frequency range greater than or equal to 10^{-5} and less than 10^{-4} occurrences per year. An event considered to be "Not Unlikely" is defined as an event with a frequency range of greater than 10^{-4} occurrences per year. Table A-4 in Appendix A of the ISA Summary provides a summary of the frequency evaluation levels used in the hazard evaluation tables.

Identified credible events can be included in the HE Tables. A "Credible" event is considered to be an event that can reasonably occur in the absence of controls. Events determined to be not credible meet one or more of the following criteria:

1. An external event for which the frequency of occurrences can conservatively be estimated as less than once in a million years ($<10^{-6}$/yr),
2. A process deviation that consists of a sequence of many unlikely events or errors for which there is no reason or motive (In determining that there is no reason for such errors, a wide range of possible motives, short of intent to cause harm, must be considered. Complete ignorance of safety procedures is possible for untrained personnel, which should be considered a credible possibility. Necessarily, no such events can ever have actually happened in any fuel cycle facility for processes similar to ACP processes), or
3. Process deviations for which there is a convincing argument, given physical laws, that they are not possible, or are extremely unlikely (The validity of the argument must not depend on any feature of the design or materials controlled by the facility's system of IROFS or management measures).

Sources of event frequency could include generic initiator database information and failure rate data from other sites (of which portions may be evaluated as applicable to ACP operations), centrifuge event history, natural phenomena frequency levels, engineering calculations, analyst judgment, and enrichment process expert opinion. The frequency level is recorded in the HE Tables in Appendix C of the ISA Summary according to the Table A-4 lettering scheme. Uncertainties in frequency levels are accommodated by erring in the conservative direction from best-estimate value. This practice is particularly important when an event frequency is just below the next highest frequency level. For example, the ISA Team considers the sources of frequency-related information, the methods used to evaluate that information, and the uncertainty associated with the evaluation process. With this information, the team might collectively decide to designate an event "Unlikely" if the event has been estimated to have an event release frequency at the high (more frequent) end of the "Highly Unlikely" frequency level.

The basis for each Unprevented Event Frequency Level listed in the HE Tables is provided in Appendix E of the ISA Summary. In general, to arrive at the unprevented frequency level for an event, a frequency for the initiator is determined through engineering judgment or by using existing applicable data when available. Then given the initiator frequency, conditional probabilities for each step in the progression to a release are estimated and combined with the initiator frequency to yield an event frequency in terms of occurrences/year. During the unmitigated phase of the HA, a control is not credited for its preventive properties when estimating the unprevented event frequency (unless the control is credited as a preventive Initial Condition in the determination of the initial unprevented frequency). If an event has multiple causes, an event frequency is developed for each cause and the cumulative event frequency is used as the overall event frequency listed in the Unprevented Frequency Level column of the table.

3.1.2.3.2.2.4.2 Natural Phenomena Hazards

For Natural Phenomena Hazard (NPH) events the severity of the design basis event (DBE) and its associated return period establish the design basis for the facility. The frequency ranges provided in Appendix A of the ISA Summary, Table A-4, are used to determine the unprevented frequency level. By design, there will be no adverse consequences to the workers or the public from a DBE. A less frequent (and more severe) event is not postulated, consistent with the philosophy that the facilities are designed to withstand the DBE. The DBE frequency for the major NPH events is provided in Table A-10 in Appendix A of the ISA Summary.

3.1.2.3.2.2.5 Unmitigated Consequence Level

Event consequences are documented by specifying the impact on the receptors. For unmitigated HA purposes, consequences are defined as the dose or exposure at specified receptor locations based upon unmitigated release of hazardous material/energy. Consequences are a function of the type and characteristics of the hazard, the quantity of hazardous material/energy released, the release mechanism, relative location of the release, and any relevant transport characteristics. Consequences are determined from (1) simple source term calculations, (2) existing safety documentation, and/or (3) qualitative assessment. The ISA Team utilizes its discretion, expertise, and knowledge of facility hazards to select one or more of the above methods appropriate for consequence determination. As in frequency evaluation, the consequence errs in the conservative direction, especially for those events with consequences at the high end of a given level. During unmitigated consequence determination, a Structure, System, and Component (SSC) or administrative control is not credited for its mitigative properties (except in those cases where the control is being credited as a mitigative IC in the determination of the initial unmitigated consequences).

Consequences are evaluated at various receptor locations to assess health effects associated with the postulated event. Table A-5 in Appendix A of the ISA Summary gives the consequence levels for radiological releases and Table A-6 provides the consequence levels for chemical releases, along with their relationship to specified receptor locations, using the maximally exposed individual at each receptor location. Appendix I of the ISA Summary presents the environmental consequences to comply with the Performance Requirements presented in 10 CFR 70.61(c)(3). The consequences presented in Tables A-5 and A-6 comply with the Performance Requirements presented in 10 CFR 70.61(b)(1-4) and 10 CFR 70.61(c)(1-4). Receptors and their locations are as follows:

Off-site Off-site receptors are the public or everyone outside the site boundary or Controlled Area. Off-site exposures are conservatively estimated (semi-quantitatively) for the public at a distance from the point of release to the nearest site boundary as follows:

Facility	**Off-site Receptor Distance in meters (ft)**
Feed and Withdrawal Building, X-3346	500 (1,640)
Feed and Product Shipping and Receiving Building, X-3346A	500 (1,640)
Interconnecting Process Piping, X-2232C	500 (1,640)
Cylinder Storage Areas – X-745G-2, X-745H, X-7746W, and X-7746S	500 (1,640)
Transportation Routes	500 (1,640)
Process Buildings, X-3001 and X-3002 (also includes Process Support Building, X-3012)	700 (2,297)
Recycle/Assembly Building, X-7725	700 (2,297)
Centrifuge Training and Test Facility, X-7726	700 (2,297)
Interplant Transfer Corridor, X-7727H	700 (2,297)
Customer Services Building, X-3344	500 (1,640)

WCA Workers in the Controlled Area are workers typically outside the restricted area, but within the controlled area of the site boundary. For evaluation purposes, these workers are located outside the last possible barrier from the hazard and at the worst possible location. Exposures are estimated (semi-quantitatively) for the WCA receptor at a distance of 100 meters (m). Typically, this would represent a point near to the exterior walls of the analyzed facility, but far enough outside that releases could have the potential to reach ground level. In general, exposures are calculated assuming exposure times are three minutes for pressurized release events, 20 minutes for fire events, and 60 minutes for slow release events.

WRA Workers in the Restricted Area are workers inside the facility. This category of receptors includes those workers in the immediate area of the hazard, and those workers in the same room or building who would quickly become aware of the hazardous condition and evacuate immediately. Exposures for the WRA are estimated qualitatively, but in all cases it is assumed that the WRA receives a dose at least as significant as the dose received by the WCA.

The Unmitigated Consequence Level column of the HE Tables indicate the estimated unmitigated impact of the release event on each of the three receptors in terms of the consequence bins of "High," "Intermediate," and "Low" as described in Table A-5 for radiological consequences and Table A-6 for chemical consequences in Appendix A of the ISA Summary.

Consequences are estimated from simple source term calculations, and/or qualitative assessment. Prior to determining the consequences of an airborne release of radionuclides, the Source Term (ST) for the radionuclides must be determined under the assumed conditions. Using the ST as input, the dose to each receptor is then determined.

3.1.2.3.2.2.5.1 Source Term Derivation

<u>Radiological Consequences</u>

In order to have conservative estimates of consequences from the accidental release of the UF_6 and UO_2F_2 inventory relating to the ACP operations, source term estimates are performed. For the type of inventory in the ACP process systems, the airborne pathway of released UF_6 and UO_2F_2 is of primary concern. The airborne source term is typically estimated by the following five-component linear equation taken from DOE-HDBK-3010-94 (Reference 7) as suggested in the *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, NUREG/CR-6410 (Reference 8).

$$\text{Source Term (ST)} = \text{MAR x DR x ARF x RF x LPF}$$

where:

MAR = Material-at Risk: amount of hazardous material available to be acted upon by a given physical stress,

DR = Damage Ratio: fraction of MAR actually impacted by the accident,

ARF = Airborne Release Fraction: the coefficient used to estimate the amount of material suspended in air as an aerosol, vapor or gas and thus available for airborne transport due to physical stress from a given accident,

RF = Respirable Fraction: fraction of airborne radionuclides or chemical aerosols that can be transported through air and inhaled into the human respiratory system, and

LPF = Leak Path Factor: fraction of radionuclides or chemical aerosols in the air transported through some confinement, deposition or filtration mechanism.

The product of the MAR x DR was conservatively determined in the unmitigated analysis on an event by event basis to estimate that quantity of the available material which could be acted upon by the event, taking into consideration the nature of the event, and the distribution of the material in the vicinity of the event. The combination of ARF and RF is selected from DOE-HDBK-3010-94 (Reference 7) based on conservative assumptions regarding the physical form of the material and the available energy during an event. The ARF and RF values depend on the event type (e.g., fire, explosion, impact, loss of confinement) and the form of the hazardous material released (e.g., predominantly UF_6 and HF gas, uranium bearing solution, and UO_2F_2 particulate). These tabulated values may be modified by calculations based on physical properties of the materials involved and the system being evaluated. A conservative value of 1.0 is typically used for the LPF in the unmitigated analysis.

The ARFs and RFs used for the consequence determination are categorized by the release mechanism and material form. The release mechanisms used are as follows:

- Fire
  - Events where the hazardous material confinement mechanism is breached by fire or is impacted by the fire.
- Explosion
  - External Explosion – Events caused by ignition of fuels or explosive gas, e.g., hydrogen generation, vehicle fuel tanks, etc.
  - Internal Explosion – Generation of explosive concentrations of flammable gases in a steel container (centrifuge casing) as a result of decomposition of contained materials due to heat, friction, etc. triggered by heat, static charge, or spark.
  - Pressurized release – Material is vented out of a container due to built up pressure.
- Loss of Containment/Confinement
  - Ambient release – Breach events with resulting release of material (e.g., leaks, etc.)
  - External Impacts/Fall – Mishandling and dropping events, impacts from external sources.

The material form during a release is:

- Predominantly Gas – UF_6 and HF from the reaction of UF_6 with moist air.
- Particulate – UO_2F_2 from the reaction of UF_6 with moist air, and UO_2F_2 stored in B-25 boxes.

- Liquid – waste containing uranium bearing solution stored in the Satellite Accumulation Areas throughout the ACP facilities.

The ARFs and RFs listed in Table 4.4-1 of the ISA Summary were taken from the DOE Handbook on Airborne Release Fractions/Rates, DOE-HDBK-3010-94 (Reference 7). The bounding release fractions were selected.

Once doses for the Public and WCA receptors are determined, these consequences are assigned as "High," "Intermediate," and "Low" according to Table A-5 in Appendix A of the ISA Summary using the radiological consequence levels for each specified receptor. For events not involving radiological consequences, the radiological consequence level is designated as "NA" (Not Applicable). The indicated consequence level bin (High, Intermediate, Low) for the WRA receptor, however, is selected qualitatively by identifying the calculated 100 m (WCA) receptor dose for each event as an initial baseline reference point. For release events, the WRA would be aware of a nearby release, as UF_6 releases are readily identified by sight, unpleasant odor, and physical discomfort if inhaled. Thus, it was assumed that the WRA would promptly relocate to avoid the release. For these events, the WRA consequence level was assumed to be equal to the WCA receptor, who is assumed to be unaware of the release.

WRA exposure equivalent to the WCA exposure is explained by using a simple expanding gas hemisphere as a release model in most cases. Assuming that the gas hemisphere radius expands at a rate of 1 m/s and the receptor walks away from the release point at 1 m/s within the cloud, it can be shown that the airborne chemical concentration levels drop off by approximately a factor of 100 within a radius of approximately 40-50 m. Workers in restricted areas could evacuate at a faster rate, putting themselves ahead of the leading edge of the expanding cloud or minimizing exposure during evacuation even if they evacuate in the direction of the plume.

Chemical Consequences and Chemical Consequence Standards

Exposure levels resulting from the accidental release of UF_6/HF were semi-quantitatively, or in the case of the WRA, qualitatively, assessed to determine airborne concentrations at each receptor. Each chemical release consequence is evaluated using the source term equation above, incorporating the same DR, ARF x RF values that were applied in the radiological consequence analysis in order to conservatively estimate the amount of UF_6/HF that becomes airborne (source term) as a result of the event. In general, the maximum off-site and on-site concentrations are then calculated by multiplying the source term by an appropriate dispersion factor (Π/Q) for the respective locations (WCA: 100 m, and Off-site: 500 m or 700 m). Similar to the radiological case above, downwind airborne concentration values for UF_6/HF releases are estimated using a Π/Q spreadsheet that calculates straight-line Gaussian plume dispersion for the receptors of interest. For the WCA, Π/Q is evaluated with a wind speed of 4.5 m/s and D atmospheric stability class. For the off-site public, Π/Q is evaluated with a wind speed of 1.0 m/s and F atmospheric stability class. Release duration depends on the nature of the event. Explosion, fire, and impact/leak events are assumed to have a 3-minute, 20-minute and 8 hour release duration, respectively. For fire events that do not involve any cylinders, the release will be assumed to occur over 20 minutes to account for the time to involve sources and breach of containment. When a cylinder is subject to fire, the internal pressure of the cylinder will build up to the rupture pressure resulting in a sudden release. In the ISA, the fire induced cylinder rupture is treated as explosion with a 3-minute release duration. The 8-hour time for impact/leak events reflects the expected conditions for low-energy steady-state releases resulting from simple breach of containment events. Although release rates varied, once the material was released from its confinement, LPFs from the building were assumed to be 1.0 for events in the unmitigated consequence analysis.

In the ISA, two simple diffusion models were developed as source term input into the straight-line Gaussian plume model spreadsheet based on a calculation for molecular diffusion from breaches in the UF_6 confinement in which no heating is involved. For releases not resulting from fire, the pre- and post-processing steps to account for plume rise and heavy gas behavior become less critical to the evaluation. The HGSYSTEM code, which is a refined Gaussian model, is not necessary to achieve the appropriate level of accuracy in this situation. Even for releases from cylinders containing liquid UF_6, the key is the size of the release relative to the surrounding atmosphere. For the liquid cylinder drop event, a flash model is developed for the evaluation of the source term. The ISA does not attempt to develop a cylinder fire model but instead uses the results from the simulation analysis used in the Cylinder Yard SAR (Reference 23). For additional detail with regard to chemical consequence determination for specific events and groups of similar events, refer to Appendix D, Event Consequence Development, of the ISA Summary.

The calculated airborne concentrations from the release and dispersion models estimated at the receptors of interest are then compared to the chemical consequence limits selected by the ISA Team. The chemical consequence limits selected are the Emergency Response Planning Guidelines (ERPGs) given in Table A-6 of Appendix A of the ISA Summary. The ERPGs are airborne concentration limits used for emergency response personnel, below which are believed that nearly all individuals could be exposed for up to one hour without experiencing certain health effects. The ERPG-1, ERPG-2, and ERPG-3 values for UF_6 are 5 mg/m^3, 15 mg/m^3, and 30 mg/m^3, respectively. Since UF_6 can readily react with the moisture in the air forming uranium compounds and HF, the chemical effects of HF have to be considered also. The ERPG-1, ERPG-2, and ERPG-3 values for HF are 1.5 mg/m^3, 16.4 mg/m^3, and 41 mg/m^3, respectively. Special ERPG values for 10-minute exposures are also used for HF, with the ERPG-1, ERPG-2, and ERPG-3 values being 1.5 mg/m^3, 41 mg/m^3, and 139 mg/m^3, respectively (Reference 9). Instead of using the ERPG values for uranium compounds, the ISA uses the uranium intakes of 10 mg, 30 mg, and 40 mg as the equivalency for ERPG-1, ERPG-2, and ERPG-3, respectively (Reference 10). From Table A.1-1 (Reference 11), the 50 percent lethality limit of soluble uranium compounds uptake is 1.63 mg U/kg body weight. With a 50 percent retention, it can be shown that the 50 percent uranium lethal intake is 228 mg for a person of 70 kg (154.4 lb). As a result, the ISA uses a 40 mg intake, which is approximately half of the 50 percent lethal intake as the equivalency of the ERPG-3. Comparison of the calculated chemical airborne concentrations at the receptor to the appropriate ERPG values (or uranium intake values) allows the assignment of a chemical consequence level of High, Intermediate, or Low to each receptor as outlined in Table A-6. For events not involving chemical consequences, the chemical consequence level is designated as "NA" (Not Applicable). Unless otherwise stated, exposures are assumed to be for one hour for all receptors and the one-hour ERPG values will be used.

High consequences for the Off-site receptor are generally based on airborne concentrations exceeding the ERPG-2 value (or 30 mg uranium intake), while Intermediate consequences to the Off-site receptor are based on exceeding the ERPG-1 value (or 10 mg uranium intake). High consequences to the WCA and WRA receptors are based on airborne concentrations exceeding the ERPG-3 value (or 40 mg uranium intake), while intermediate consequences to the WCA and WRA receptors are based on concentrations exceeding the ERPG-2 value (or 30 mg uranium intake). For those events that involve only the release of UF_6 from cylinders or pipes in the absence of fire, the rate of diffusion of UF_6 is generally very low such that the UF_6 has sufficient time to react with air and the product UO_2F_2 has time to deposit or plate out. Only the peak HF concentrations are used to compare with the ERPG values for both on-site and off-site receptors during these events. The consequence classification for HF is based upon the peak HF concentration at any time during the event.

Environmental Consequences

Environmental consequences were addressed by the ISA Team when considering the credible accident scenarios where release quantities exceeded the levels established by the Performance Requirements of 10 CFR 70.61(c)(3). The methods used and results are provided in Appendix I of the ISA Summary.

3.1.2.3.2.2.6 Unmitigated Risk Level

Using event frequency and consequence levels, the events are "binned" in frequency-consequence space to assess relative risk in accordance with 10 CFR 70.61. A risk rank for each receptor is individually determined for both radiological consequences and chemical consequences. The objective of risk binning is to focus attention on those events that pose the greatest risk to the public and workers. Higher risk events are candidates for additional analysis and/or selection of IROFS to reduce the risk.

Tables A-7, A-8, and A-9 in Appendix A of the ISA Summary are risk binning matrices for the three receptor locations considered in the ISA [i.e., WRA (close-in), WCA (100 m), and Off-site (500 m or 700 m)]. Table A-7 is the risk binning matrix for the Worker in the Restricted Area, who is typically located anywhere inside the facility with the hazardous release or hazardous condition. Table A-8 is the risk binning matrix for the Worker in the Controlled Area (100 m receptor) located outside the facility. Table A-9 is the risk binning matrix for off-site receptors (Public).

In each of these tables, a rectangular matrix defines bins in frequency-consequence space. Each bin that is lettered with the letter "A" indicates that 10 CFR 70.61 Performance Requirements are exceeded, in which case IROFS must be implemented to reduce the risk. Alternately, bins designated with the letter "B" indicates that 10 CFR 70.61 Performance Requirements are met, and no IROFS are required.

Accidents that are considered not to be "Credible" are generally not shown, but would have a risk rank of "B." Accidents that have Low consequences have a risk rank of "B." In either case, the risk rank of "B" requires no further analysis or designation of IROFS to control risk (unless the control is an IC, in which case the control would be designated as an IROFS).

The HE Tables in Appendix C of the ISA Summary provide a bin letter in the unmitigated risk level column for both radiological and chemical consequences, representing risk for each receptor location for each of the postulated events.

3.1.2.3.2.3 Available Preventive and Mitigative Controls

3.1.2.3.2.3.1 Preventive Controls

A preventive control is any feature that may be relied upon to reduce the frequency of a hazardous event (up to the point of release of hazardous material/energy). The selection of preventive controls is made without regard to any possible pedigree of the feature such as procurement level or current classification. Preventive controls might include engineered features (e.g., SSCs), administrative controls (e.g., operator actions), natural forces or physical phenomena (e.g., ambient conditions, buoyancy, gravity), or inherent features (e.g., physical or chemical properties, location, elevation) operating individually or in combination. Controls that could serve preventive functions are listed in the Preventive Controls column of the HE Tables, and are sub-divided into administrative and engineered (design) controls for each event. It is from this list that the controls needed to prevent hazardous events are selected. The ISA Team utilize this list to select and subsequently credit preventive controls as IROFS to reduce the frequency of the postulated release events. The prevented event frequency as given for a particular event takes into account any credited (bolded) preventive controls (preventive IROFS) in the HE Tables which act to reduce the frequency of the event (i.e., to reduce the frequency of the initiator and/or to reduce the probability of the progression of occurrences which ultimately lead to the release of hazardous material/energy).

3.1.2.3.2.3.2 Mitigative Controls

Mitigative controls are any features that could reduce the consequences associated with the release of hazardous material/energy. The identification of such controls is made without regard to any possible pedigree of the feature such as procurement level or current classification. Mitigative controls are those that are assumed to be operable during an event or post event, and are not required to be operating prior to the event initiation. Therefore, mitigative controls must be capable of withstanding the environment of the event. These might include engineered features (e.g., SSCs, detection systems), administrative controls (e.g., operator actions), natural forces or physical phenomena (e.g., ambient conditions, buoyancy, gravity), or inherent features (e.g., physical or chemical properties, location, elevation) operating individually or in combination. Controls that could serve mitigative functions are listed in the Mitigative Controls column of the HE Tables, and are sub-divided into administrative and engineered (design) controls for each event. It is from this list that the controls needed to mitigate hazardous events are selected. The ISA Team utilize this list to select and subsequently credit mitigative controls (mitigative IROFS) to either reduce the material released once a release occurs, or reduce the consequences of the release event to the receptors of interest.

3.1.2.3.2.3.3 Subdivision of Preventive and Mitigative Controls

Preventive and mitigative controls can be subdivided into active engineered controls, passive engineered controls, and administrative controls. Active engineered controls are physical devices that use active sensors, electrical components, or moving parts to maintain safe process conditions without any required human action. Passive engineered controls are devices that use only fixed physical design features to maintain safe process conditions without any required human action. Administrative controls are procedurally required or prohibited actions, combined with or without a physical device that alerts the operator that the action is needed to maintain safe process conditions, or otherwise adds substantial assurance of the required human performance.

3.1.2.3.2.4 Control Selection and Mitigated Hazard Evaluation Development

Following the Unmitigated Hazards Evaluation step, controls were identified using the methodology given in NUREG-1520 (Reference 3) for designation as IROFS. The controls selected as IROFS are necessary to bring the risk of unprevented and unmitigated accidents to within the Performance Requirements of 10 CFR 70.61, or to capture Initial Conditions that were established in the unmitigated Hazards Analysis as safety basis controls. Controls include engineered controls such as SSCs and also administrative controls or programs that provide a safety function. Defense in Depth (DID) concepts utilizing non-credited controls were also incorporated into the control strategy for a postulated event whenever possible.

3.1.2.3.2.4.1 Control Selection Method

First, candidate non-credited controls for each postulated event are listed in the Preventive Controls Column and Mitigative Controls Column of the HE Tables in Appendix C. The candidate controls for each event can then be either: 1) credited as IROFS, if necessary, to prevent or mitigate a release event, or 2) remain non-credited controls, which are available to provide DID, but which require no control "pedigree." For those events in which the unmitigated risk exceeds Performance Requirements of 10 CFR 70.61, appropriate controls are required to be selected from the candidate controls and credited as IROFS in preventing and/or mitigating the subject event until the mitigated risk is within the Performance Requirements. Other controls which exist but which are not selected and designated as IROFS, provide a DID function.

The unprevented frequency and unmitigated consequences of each event are compared with the 10 CFR 70.61 Performance Requirements for each receptor. These Performance Requirements for each of the three receptors (WRA, WCA, and Off-site) are presented in Tables A-7, A-8, and A-9 in Appendix A of the ISA Summary. Those unmitigated events whose risk exceeded the 10 CFR 70.61 Performance Requirements were marked for control selection to reduce the event frequency or mitigate the event consequences to within the Performance Requirements. Preventive controls that were credited for reducing the frequency in the Mitigated

HA columns are set in bold font type followed by IROFS numbers in the HE Tables Preventive Controls column and are also provided in the List of IROFS in Section 7.2 of the ISA Summary. The prevented event frequency given for a particular event takes into account any credited (bolded) preventive controls in the HE Tables, which act to reduce the frequency of the event. Preventive controls not explicitly credited in this way to reduce frequency provide DID. Similarly, mitigative controls that were credited in mitigating consequences are set in bold font type followed by IROFS numbers in the HE Tables Mitigative Controls column and are also provided in the List of IROFS in Section 7.2 of the ISA Summary. The mitigated consequences estimated for a particular event takes into account any credited (bolded) mitigative controls in the HE Tables which act to reduce the severity, material released, or dose (or chemical exposure) due to the event.

Table F-1 in Appendix F of the ISA Summary, a control selection table for risk reduction, was developed by the team for each unmitigated event with risk exceeding the established Performance Requirements to record the process of selecting controls that would reduce the frequency of, and/or lessen the severity of, each applicable event to within the Performance Requirements. The table presents the credited risk reduction to the applicable receptors for each credited control (i.e., IROFS). Estimated frequency reduction values for each credited preventive IROFS were given to arrive at a "prevented" event frequency for each event cause. Similarly, estimated consequence (dose or chemical exposure) reduction values for each credited mitigative IROFS were presented to arrive at a mitigated consequence for each receptor. The prevented frequency and the reduced consequence level for receptors that did not require controls (i.e., those receptors with an unmitigated risk in the "B" risk bin) are designated as "NA."

3.1.2.3.2.4.2 Control Selection Preference

In general, controls were selected using an order of preference. The first controls credited were the "see and flee" controls, which include Emergency Response Actions; Alert, Notification, and Protective Actions; and Trained Operator Actions. These controls are credited with reducing potential radiological and chemical consequences to all receptors. These controls were applied first, as crediting receptors with minimizing their exposure to a hazardous chemical release is a control of very high reliability. Then, additional controls were applied, as necessary, with preference given to certain types of controls over other types of controls. In general, available preventive controls were generally selected before additional mitigative controls so as to prevent or reduce the frequency of the event rather than attempt to mitigate the event consequences after the event has occurred. If available, engineered or designed controls were selected before administrative controls to utilize the inherent reliability advantage of designed systems or components over that of required human action compliance. In the case of engineered controls, where possible, passive engineered controls were generally selected before active engineered controls due to the increased reliability of a passive engineered feature. Factors such as reliability, durability, life cycle cost, facility operating life, applicability to multiple events, etc. were also considered during control selection and had some influence on the preferred selection strategy.

3.1.2.3.2.4.3 Preventive or Mitigative Value of Control

While it is often difficult to estimate the value of a specific control in providing event frequency reduction or consequence mitigation, several general guidelines were used to assist in control value estimation, in the absence of more detailed information.

3.1.2.3.2.4.3.1 Preventive Control Value

With regard to preventive controls, a passive engineered control (such as a nozzle or orifice in limiting flow, or a concrete jersey barrier for limiting vehicle access or impacts) would typically be credited as providing a frequency reduction of three orders of magnitude (frequency may be reduced by 1×10^{-3}). An active engineered control (such as negative pressure ventilation system, an automatic valve or an automatic fire suppression system) would be credited as providing a frequency reduction of two orders of magnitude (frequency may be reduced by 1×10^{-2}). An administrative control (such as operator actions) would typically be credited as providing a frequency reduction of only one order of magnitude (reduced by 1×10^{-1}) due to the potential for human error. These values are supported by, and are generally more conservative than the example control values outlined in Table A-10 of Appendix A of the ISA Summary as compared to Chapter 3 of NUREG-1520 (Reference 3). It should be noted that these are general preventive control values that the ISA Team considered as a starting point. Any vulnerabilities or strengths in a particular control could be reason for the team to vary the general value of these types of controls for the specific situations involved in a particular event.

3.1.2.3.2.4.3.2 Mitigative Control Value

Mitigative controls reduce either the amount of material released, or the potential dose or airborne chemical concentration to a receptor attributed to the release. The value of the mitigative control varies with the effectiveness of the control with relation to the nature and energy of the release event. For instance, the value of certain mitigative controls (e.g., HEPA filtration) may be fairly easy to quantify. As a general example, HEPA filtration incorporates an engineered efficiency of approximately 99.9 percent, and therefore may be confidently considered to reduce the dose to an external receptor by three orders of magnitude (dose reduction by approximately 1,000) due to the efficiency of the filtration mechanism (given that the released hazardous material, in fact, follows the filtered release path and the filter survives the event intact). In some events, a mitigative control such as a centrifuge casing was credited with sufficient confinement capability relative to the nature of the event, so as to limit the subsequent doses to receptors.

However, the determination of the mitigative value of an administrative control such as worker evacuation from the immediate scene of an unfiltered radiological or chemical release is more subjective and difficult to quantify. The ACP utilizes a "See and Flee" policy to protect the health and safety of workers who may encounter a release of UF_6 or other hazardous material. The policy is for employees to promptly move to a safe location away from the immediate release area. The "See and Flee" policy has been utilized effectively at the gaseous diffusion plants for numerous years, in conjunction with other plant programs/controls, in limiting exposures to plant workers to safe levels (thousands of hours of operation with hundreds of thousands of pounds of in-process UF_6 at pressures much greater than the pressures in the ACP). The results have been minimal exposure to workers, even from a sizable release. In addition, experience indicates that workers can readily recognize even incidental releases of UF_6 and take appropriate actions to evacuate the area of the release. "See and Flee" is credited with mitigative values on a case-by-case basis, with appropriate consideration that the worker in the vicinity of the release has the ability to evacuate due to the conditions likely to be present during the postulated accident scenarios. In general for this analysis, the worker's ability to recognize a radiological or chemical upset condition and immediately evacuate the area was qualitatively estimated to reduce the dose to the worker by a range of approximately two to three orders (1/100 to 1/1,000) of magnitude. This value is subjective and may vary on a case-by-case basis depending on the nature and rapidity of the event, worker awareness, available egress routes, and the ability and time to take protective action (evacuation). In general, the ISA Team considered that WCA protective actions were also worth approximately two orders of magnitude (1/100) consequence reduction, again subject to specific event conditions. For the Off-site Public, the mitigative control of alert/notification and sheltering/evacuation was deemed by the ISA Team to result in a conservative consequence reduction of only one order of magnitude (1/10), in that the response of the public is considered to be less reliable than that of trained site workers. Refer to Tables F-1 through F-11 and the associated text in Appendix F of the ISA Summary for the values assigned to each credited preventive and mitigative IROFS for each event cause and receptor.

Controls were required to be credited in all events for which the unmitigated risk exceeded 10 CFR 70.61 performance requirements. In addition, for certain events (including events whose unmitigated risk did not exceed performance requirements), Initial Conditions may have been credited inherently in the unprevented frequency and unmitigated consequences for certain events, by initially limiting the frequency or consequences of the event. For example, for the massive river flooding event, the location and elevation of the site well above the Maximum Probable Flood crest level was credited as an initial condition in establishing the unprevented frequency for the event in the "Highly Unlikely" frequency level. The team would look for and capture these types of Initial Conditions as an inherent credited control (an IROFS) for that event, regardless as to whether the unmitigated risk associated with the event exceeded Performance Requirements.

3.1.2.3.2.4.4 Control Selection Results

The credited controls identified for each event were grouped and consolidated, and are presented in Table 7.2-1 of the ISA Summary, including controls credited as initial conditions. Table 7.2-1 presents grouped controls under an appropriate Control Strategy heading, whether the control constitutes a design feature, or an administrative control, and the applicable event(s) from the HE Tables in Appendix C of the ISA Summary to which the control applies. A description of each credited control (i.e., IROFS) is also given in Chapter 7.0 of the ISA Summary including the safety function and credited attributes of the control. IROFS are also denoted by controls listed in bold type followed by IROFS numbers in the Preventive and Mitigative Controls column of the HE Tables in Appendix C of the ISA Summary. As previously noted, the preventive and mitigative reduction values of these IROFS are presented in

Tables F-1 through F-11 and the associated text of Appendix F of the ISA Summary for each event.

3.1.2.3.2.4.5 Implementation of Controls

Procedural IROFS listed in Table 7.2-1 of the ISA Summary and IROFS which involve operation of equipment to perform the safety function, also require associated training conducted to familiarize Workers with the procedure and/or equipment. In addition, for each SSC credited as an IROFS, periodic surveillances (inspections) and preventive maintenance should be developed for the SSC during implementation, as validation of the operability of the SSC. Other general programmatic controls such as facility configuration control and inventory control are not specifically identified or credited as an IROFS for each event, although implementation of these controls is assumed to maintain the continuing validity of the IROFS.

3.1.2.3.2.5 Mitigated Risk Level

Once the prevented event frequency and mitigated consequence levels are determined from the crediting of IROFS, the events are risk-binned again in frequency-consequence space to assess the mitigated risk relative to 10 CFR 70.61 Performance Requirements. Similar to the unmitigated analysis, Tables A-7, A-8, and A-9 are also used as the risk binning matrices for the mitigated risk comparison for each receptor (WRA, WCA, and Off-site, respectively). Following the crediting of IROFS, the mitigated risk for the event is expected to fall in a bin designated "B," indicating the Performance Requirements have been met. If the mitigated risk bin remains within the "A" designation indicating the Performance Requirements are still exceeded, then either additional analysis must be performed, or additional IROFS must be identified and credited. The mitigated risk level for receptors that did not require controls (i.e., those receptors with an unmitigated risk in the "B" risk bin) is designated as "NA." While not preferred, in the event that no additional IROFS are available or no more refinement is to be gained from any additional analysis that might confirm a reduced risk when compared to that previously estimated in the unmitigated Hazard Evaluation, then the NRC may at their discretion, consider acceptance of a "Residual Risk" from the event to Workers or to the Public.

3.1.2.3.2.6 Evaluation of Mitigative IROFS Failure

A consideration in the identification of mitigative IROFS is the possibility that these controls could fail to perform their safety functions. Given this possibility, events for which mitigative controls were credited were evaluated to examine the residual risk associated with the postulated failure upon demand of each mitigative IROFS. The approach used in this evaluation develops a series of sub-events designed to demonstrate that the risk of the event following failure of one or more of the credited mitigative controls is still within the 10 CFR 70.61 Performance Requirements. This evaluation is summarized in Appendix K of the ISA Summary.

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

3.1.2.3.2.7 Evaluation of Criticality Events for Commercial ACP Operation

The methodology utilized for evaluating criticality events for the commercial ACP operations (i.e., non-HALEU) is described in this section. The method for evaluating criticality events for HALEU Demonstration is described in Section 3.1.2.3.2.8. Additionally, changes to criticality accident sequences for commercial plant (i.e., non-HALEU ACP) will be performed using the methodology provided in Section 3.1.2.3.8.

Criticality Events are derived and evaluated in a similar manner as radiological and chemical release events are revised and evaluated. Reviews are conducted of the ACP facilities and operations to determine the hazards that are present then further review is conducted to determine the credible accident sequences. The credible accident sequences are evaluated to determine the potential consequences and the frequency with which the accident sequences could occur assuming no controls. Criticality events are assumed to have high consequences in a localized area, so they must be made "Highly Unlikely." (For criticality events, since the consequences only take place in a localized area (well under 100 meter distance), the dose received by the WRA is assumed to be High and the dose expected for the WCA and the Off-site public is assumed to be Low.) No mitigative controls are available to reduce the assumed high consequences to within the 10 CFR 70.61 Performance Requirements.

In addition to the requirement to make high consequence events "Highly Unlikely," criticality events must have double contingency controls. For the initial ACP ISA effort, Nuclear Criticality Safety (NCS) Reports were generated to document the NCS analysis of the general ACP facilities and operations. The NCS Reports identified "What-If" events to assist in the establishment of double contingency controls as required by 10 CFR 70.24.

A review of the NCS Reports was conducted and documented within an Engineering Evaluation (Reference 15) to ensure the "What-If" events were adequately addressed by criticality event sequences. Those "What-If" events determined not to credibly contribute to a criticality event were documented as such. Those "What-If" events determined to credibly contribute to a criticality event were documented in the ISA and evaluated to ensure the frequency of the associated criticality event was "Highly Unlikely" by identifying appropriate IROFS as necessary. Release events that could lead to a subsequent criticality that have been made "Highly Unlikely" due to chemical consequences require no further analysis for subsequent criticality concerns, as the initiating release is already "Highly Unlikely."

As the ACP design is finalized, NCS Evaluations (NCSEs) will be generated to document the NCS analysis of the specific ACP facilities and operations. Similar to the review performed on the NCS Reports, a review of the NCSEs will be conducted and documented to ensure the NCSE "What-If" events are adequately addressed by criticality event sequences. The NCSEs will be reviewed to ensure agreement with the ISA. Any required ISA changes will be processed in accordance with 10 CFR 70.72 requirements.

Finally, consideration for chemical release events was made to address the large release events that were mitigated to be "Low" consequences, but could still release hazardous material in quantities that exceed the minimum critical mass (20 kg UF_6 at 10 wt. percent ^{235}U per Reference 16). Appropriate additional controls were credited as necessary to ensure a subsequent criticality to those release events was "Highly Unlikely."

3.1.2.3.2.8 Evaluation of Criticality Events for HALEU Demonstration

The method for evaluating criticality events for HALEU Demonstration is described in this section, in conjunction with the following aspects of Section 3.1.2.3.2, "Hazard Evaluation," of the ISA Summary that apply to both criticality and non-criticality events: (1) the use of initial conditions from Section 3.1.2.3.2.1, (2) the criteria for events that are considered "Credible" from Section 3.1.2.3.2.2.4.1, and (3) consideration of Natural Phenomena Hazards from Section 3.1.2.3.2.2.4.2. Other aspects of the methods described in this and other portions of Section 3.1.2.3.2 of the ISA Summary do not apply. With regard to consequence, criticality is presumed to be "high consequence." Since the consequences only take place in a localized area (well under 100 meter distance), the dose received by the WRA is assumed to be High and the dose expected for the WCA and the Off-site public is assumed to be Low. Mitigative controls are not applied. The method used for hazard evaluation of criticality events is described below.

The evaluation of HALEU Demonstration Criticality Events was performed in accordance with the deterministic, parameter-based approach of NUREG-1520, Chapter 5, Appendix C, "Example Procedure for Subcriticality Evaluation." This method demonstrates compliance with the requirement of 10 CFR 70.61(d) to ensure that, under normal and credible abnormal conditions, all nuclear processes are subcritical, including an approved margin of subcriticality for safety. As stated in NUREG-1520, Chapter 5, Appendix A, "Nuclear Criticality Safety Performance Requirements and Double-Contingency Principle" (DCP), 70.61(d) is more restrictive than 70.61(b), and "if one meets § 70.61(d), then one also automatically meets § 70.61(b)." Whereas "the spectrum of credible abnormal conditions in 10 CFR 70.61(d) need not consider upsets beyond those required for compliance with the double contingency principle", "adherence to the DCP can be one means of meeting the performance requirements of § 70.61(d) (and therefore also § 70.61(b))." This deterministic approach of NUREG-1520, Chapter 5, Appendix C was selected for evaluating criticality hazards in the HALEU Demonstration ISA because it is based on the traditional, time-tested approach to NCS as endorsed in Chapter 5 of NUREG-1520, with its long track record of safety in the nuclear fuel industry.

Criticality Events for the HALEU Demonstration Project were derived and evaluated through the process of generating Nuclear Criticality Safety Evaluations (NCSEs). The NCSEs were developed using a parameter-based method that begins with "a consideration of normal and abnormal conditions." Such an approach provides assurance that all conditions that can lead to an inadvertent criticality are identified. Controlled parameters, and limits on those parameters, are identified to ensure subcriticality. The specific controls with the safety function of maintaining controlled parameters within their safety limits are documented in NCSEs. Systems of controls which together perform the same safety function (i.e., maintain a particular safety limit) may be grouped together in items relied on for safety (IROFS). ~~Failure of an IROFS is considered to have occurred when it fails to perform its safety function (*i.e.*, when the associated safety limit is exceeded).~~

Demonstration of subcriticality under 10 CFR 70.61(d) is done through means of compliance with the DCP which requires at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. The following guidance is provided on the various terms in the definition of the DCP.

- *Unlikely* changes in process conditions should be expected to occur rarely, or not at all, during the lifetime of the facility. Operational events that occur regularly should not be credited as a contingency relied on to meet the DCP (although they may constitute part of a contingency if a combination of events may be considered unlikely).

- *Independent* changes in process conditions are such that one contingency neither causes another contingency nor increases its likelihood of occurrence. The existence of any credible common-mode failure of both contingencies means that it is not valid to consider them independent. Therefore, independent changes in process conditions are ensured by following the preference for control of diverse parameters or, when relying on single-parameter control, demonstrating the lack of any credible common-mode failure.

- *Concurrent* does not mean that the two changes in process conditions must occur simultaneously, but that the effect of the first contingency persists until the second contingency occurs. Therefore, concurrence of changes in process conditions is addressed by providing means for prompt detection and correction of abnormal conditions (e.g., periodic surveillance, process monitoring).

- *Changes in process conditions* do not imply that reliance on two different parameters is mandatory to satisfy the Double Contingency Principle. Reliance on two different parameters is preferred over reliance on multiple controls on a single parameter. It is difficult to achieve complete independence when controlling one parameter. In those cases in which single parameter control cannot be avoided, the analysis in the applicable NCSE will ensure and document that no common-mode failures exist.

As stated in NUREG-1520, Chapter 5, Appendix A, the DCP is sufficient for satisfying the 70.61 performance requirements provided the following additional conditions are met:

1. Controls are established on system parameters to preclude changes in process conditions, and these controls are designated as IROFS;

2. The condition resulting from the failure of a leg of double contingency has been shown to be subcritical with an acceptable margin; and

3. Controls are sufficiently reliable to ensure that each change in process conditions necessary for criticality is "unlikely." Management measures are established to ensure they are available and reliable to perform their safety function.

To provide additional guidance for satisfying the criteria discussed above, NUREG-1520, Chapter 5, Appendix A contains several examples of scenarios implementing the Double Contingency Principle that are stated as satisfying the performance requirements of 10 CFR 70.61. For scenarios that can be shown to satisfy the Control Sets below, no additional justification is needed for why the performance requirements of 10 CFR 70.61 are satisfied.

Control Set A: *A passive geometry control in which no credible failure mode (e.g., bulging, corrosion, or leakage) exists and which has been placed under configuration management.* An example scenario consistent with this definition is a favorable geometry vessel in a benign environment for which corrosion or degradation is not credible, vessel construction is so robust that a leak is not credible, and there is no credible means for the material to accumulate in an unfavorable configuration.

Control Set B: *Two passive controls in which there is a credible failure mode, and there are sufficient management measures to ensure the controls continue to perform their safety functions (e.g., periodic surveillance to detect corrosion/bulging).* An example scenario consistent with this definition is a storage array in which fissile material is stored in fixed geometry containers, and the spacing between containers is provided by fixed devices, with geometry and spacing controls ensured by the configuration management program and by periodic walkthroughs of the storage array process area.

Control Set C: *One passive control under configuration management and one active engineered control whose reliability is ensured by periodic functional testing, maintenance, and an alarm to automatically indicate its failure.* An example scenario consistent with this definition is a calciner relying on geometry and moderation control in which geometry control is provided by limiting the calciner interior to the height of a single layer of fissile material boats, and moderation control is provided by monitoring of the calciner temperature. Temperature control is ensured by thermocouples that alarm if the temperature drops below a minimum set-point.

Control Set D: *One engineered and one enhanced administrative control in which the instrumentation and devices included in the administrative control are subject to periodic functional testing and maintenance, and the operator action is performed routinely or reinforced by periodic drills and training.* An example scenario consistent with this definition is a vessel in which the volume of fissile solution is controlled by the diameter of the tank and by procedurally limiting the solution height. In addition, the operator actions are supported with a high-level switch equipped with an alarm.

Control Set E: *One engineered control and one simple administrative control in which the reliability of the administrative control is subject to a high degree of redundancy.* An example scenario consistent with this definition is a solution transfer from favorable to unfavorable geometry relying on two controls on concentration. Two different operators are required to draw separate samples which are then analyzed in the laboratory by two different methods and shown to be within concentration limits before transfer is authorized. In addition, the area supervisor maintains control of a key to the transfer pump so that the procedure may not be inadvertently bypassed. These operator actions are backed up with an in-line sodium iodide detector that automatically closes an isolation valve if concentration limits are exceeded.

Control Set F: *Two administrative controls that are independent (e.g., performed by different individuals or verified by a supervisor), for which human factors have been considered in the design of the process such that the operation is not prone to error, and there is sufficient margin to require multiple failures before the criticality control limit can be exceeded.* An example scenario consistent with this definition is a glovebox relying on dual mass control in which two operators or an operator and a supervisor must confirm that placing material into the glovebox will not result in the mass limit being exceeded. In addition, criticality would require the mass limit to be exceeded multiple times, which would be difficult to achieve and would be readily apparent.

The Control Set being referenced as a basis for satisfying the Double Contingency Principle for a given HALEU Demonstration Criticality Event is documented in the appropriate Double Contingency Evaluation Table for HALEU Demonstration Criticality Events, contained in Appendix C of LA-3605-0003A, *Addendum 1 of the Integrated Safety Analysis Summary for the American Centrifuge Plant – HALEU Demonstration* (Reference 21). Additional justification is provided for any scenario that does not fall into one of the above Control Sets (e.g., by ensuring there is no credible event leading to criticality, or by crediting natural and credible course of events). An example of this type of scenario is a facility storing contaminated soil or equipment with a very low uranium concentration in which there is no known concentration mechanism that can lead to a critical configuration.

The Control Sets satisfy the requirements of the Double Contingency Principle and the performance requirements of 10 CFR 70.61 (b) and (d), and are summarized in Table A-1 of Appendix A of LA-3605-0003A (Reference 21).

3.1.3 Management Measures

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

3.2 Integrated Safety Analysis Summary

An ISA Summary for the ACP (Reference 1) and *Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration* (Reference 21), meeting the requirements of 10 CFR 70.65(b) was prepared in accordance with the guidance contained in Chapters 3.0 and 5.0 of NUREG-

1520, *Standard Review Plan for the Review Fuel Cycle Facilities License Applications,* and NUREG-1513, *Integrated Safety Analysis Guidance Document.* The ISA Summary is being submitted for review (separate from this license application).

3.3 Items Relied on For Safety Boundary Definition

In order to ensure IROFS are available and reliable, their boundaries must be clearly established. The IROFS boundary determination process relies upon the ISA to identify and define the IROFS and their functions. The boundary determination process then uses the ISA and ACP design documentation to establish and identify what structures, systems, components, and actions are required to fulfill the IROFS functions. IROFS boundaries are defined using CMP-3601-0001, "IROFS Boundary Determination Plan."

3.4 Seismic and Wind Specifications

Seismic specifications for the ACP design are based on the risks and potential consequences from seismic events involving the primary facilities. This approach results in two criteria being applied depending upon whether or not the normal operations therein involve liquid UF_6. Facilities where liquid UF_6 operations occur (non-HALEU, commercial ACP operations only) are required to withstand the forces resulting from a 10,000-year return period seismic event. All other facilities (including both non-HALEU commercial ACP operations and the HALEU Demonstration) are required to withstand the forces resulting from a 1,000-year return period seismic event because UF_6 operations therein involve UF_6 in either gas or solid form.

The X-3344 Customer Services Building (used in non-HALEU commercial ACP operations only) is designed to withstand a 10,000-year return period seismic event for the Piketon, Ohio area. This correlates to a conservative assumption of 0.48 gravity Peak Ground Acceleration (PGA) (Reference 13). The corresponding vertical earthquake ground motion is two-thirds of the horizontal ground motion or 0.32 gravity PGA. These PGA values are based on earlier geotechnical studies (References 13, 17, and 18). The results of these studies are documented and summarized in EE-3100-0003, *Summary of ACP Seismic Design Values* (Reference 19).

The X-2232C Interconnecting Process Piping; X-3001 and X-3002 Process Buildings; X-3012 Process Support Building; X-3346 Feed and Withdrawal Building; X-3346A Feed and Product Shipping and Receiving Building; X-7725 Recycle/Assembly Building; X-7726 Centrifuge Training and Test Facility; and X-7727H Interplant Transfer Corridor are designed to withstand a 1,000-year return period seismic event for the Piketon, Ohio area. This correlates to a conservative assumption of 0.15 gravity PGA (Reference 12). The corresponding vertical earthquake ground motion is 0.1 gravity PGA.

IROFS structures, systems, and components required to function in response to seismic events are constructed and/or installed to withstand the forces stated above. Non-IROFS structures, systems, and components are constructed and/or installed, as necessary, to ensure they cannot adversely affect IROFS structures, systems, and components.

Seismic response spectra for the ACP are documented in EE-3100-0003, *Summary of ACP Seismic Design Values* (Reference 19). The 10,000-year response spectrum identified in the summary has been used to perform dynamic analyses of the X-3344 to ensure it can withstand a 10,000-year return period event. The 1,000-year response spectrum identified in the summary has been or will be used to perform dynamic analyses of the X-2232C, X-3001 X-3002, X-3346, and X-3346A to ensure they can withstand a 1,000-year return period event. Dynamic analyses of the X-3012, X-7725, X-7726, and X-7727H were performed as part of the original plant design to ensure their design integrity using the original seismic response spectrum associated with a 1,000-year return period event (Reference 12). It was deemed unnecessary to repeat these analyses because the ACP is not changing the design or installed configuration of these facilities and the response spectrum used in the original analysis (Reference 2) adequately bounds the current response spectra derived from more recent geotechnical studies (Reference 13, 17 and 18). A comparison of the original response spectrum to the current response spectrum is documented in EE-3901-0004 *Dynamic Analysis Verification on Existing ACP Buildings* (Reference 20). These analyses ensure that the primary facilities are adequately designed to prevent collapse of the structures during major seismic events and ensure the subsequent release of licensed material in a manner that could cause the 10 CFR 70.61 Performance Requirements to be exceeded is highly unlikely. All other process support or process related buildings or structures will be designed or have been previously designed for a 1,000-year return period event. Non-IROFS structures have been or will be designed using regional building code values.

The original PGA listed in ORO-EP-120 (Reference 12) for a 1,000-year event is 0.15g. This PGA value is the same as used in the 1982 Beavers study (Reference 14), the 1995 three-site seismic study that included the Portsmouth reservation (Reference 22), and the current ACP seismic design criteria (Reference 19). There are minor differences in the response spectra for the ACP.

In addition, wind specifications for the ACP design are also based on the risks and potential consequences from high wind events involving the primary facilities. These include a tornado and events that involve straight winds that have sufficient velocity to cause physical damage to site buildings and HALEU Demonstration areas. The frequency that a tornado would strike the HALEU Demonstration areas and cause a release of hazardous material has been evaluated as "Highly Unlikely." It is estimated that a tornado striking any HALEU Demonstration area(s) would result in "Low" radiological and chemical consequences for all receptors. Because of the "Highly Unlikely" frequency assigned to the tornado events, the unmitigated risk associated with the event does not exceed Performance Requirements; therefore, no controls are required to be imposed.

The wind loading design basis criteria for the Process Building X-3001 associated with the HALEU Demonstration is 100 mph straight-line winds at a 20,000 year return period, and for the Interplant Transfer X-7727H Corridor, X-7725 Building, and X-7726 Facility is 90 mph straight-line winds at a 3,000 year return period. Therefore, no release of hazardous material would be expected from these buildings during a design basis straight-line wind event. The unprevented frequency of a design basis straight-line wind event impacting these buildings is evaluated as "Unlikely" for the Process Building and "Not Unlikely" for the X-7727H Corridor, X-7725 Building, and X-7726 Facility.

The wind design loads used in the original design of the existing structures was prescribed in K-DA-603 (reference 24). The wind design load specifications included the original design document K-DA-603 were referenced in determining the loads used for new structures. The basis for the resulting wind criteria to be utilized in new designs was captured in Project Specification 662574.215.000910 (reference 25).

3.5 Integrated Safety Analysis Maintenance

As stated previously, the ISA is a compilation of the design and analysis documentation utilized to identify the potential accident sequences that could occur, designate IROFS to either prevent such accidents or mitigate their consequences to an acceptable level, and identify the management measures to provide reasonable assurance of the availability and reliability of IROFS. The ISA Summary is a synopsis of the ISA and contains the information required by 10 CFR 70.65(b). The ISA Summary is updated to reflect changes to the ISA.

The ISA accounts for any changes made to the ACP facilities or its operations are evaluated in accordance with the requirements of the 10 CFR 70.72 change process. Any facility change, operational change, or change in the process safety information that may alter the parameters of an accident sequence is evaluated by means of the ISA methods. The Licensee periodically reviews IROFS per the requirements of 10 CFR 70.62(a)(3) to ensure their availability and reliability for use and consistency with the ISA. The Licensee evaluates whether changes to existing IROFS and associated management measures are required, or if new IROFS or management measures are required. The bases (including assumptions and initial conditions) for the ISA are maintained and controlled via the various management measures identified in Chapter 11.0 of this license application. This includes, but is not limited to the preventive maintenance, corrective action, configuration management, and audit/assessment programs.

3.6 References

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3. NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications*, U. S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Washington, DC, Revision 2

4. 40 CFR Part 68, *Chemical Accidental Prevention Provisions*, U. S. Environmental Protection Agency, Washington, DC

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

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

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

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15. E. L. Pyzik and G. S. Corzine, *Nuclear Criticality Safety Reports "What-if..." Hazard Analysis Review for the American Centrifuge Plant*, EE-3601-0014, Rev. 2, March 2008

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18. *Geotechnical Investigation – American Centrifuge Plant*, Project No. FACP-2063, Prepared by Fugro, William, Lettis and Associates Inc., June 2010

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20. Hortel J.M. and Corzine G.S., *Dynamic Analysis Verification on Existing ACP Buildings*, EE-3100-0004, Revision 0, October 2013

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22. ES/CNPE-95/2, *Seismic Hazard for the Oak Ridge, Tennessee; Paducah, Kentucky; and Portsmouth, Ohio; U.S. Department of Energy Reservations,* Center for Natural Phenomena Engineering, Lockheed Martin Energy Systems, Oak Ridge, TN; December 1995

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4.0 RADIATION PROTECTION

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

4.1 Radiation Protection Program Implementation

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

4.2 As Low As Reasonably Achievable Program

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

- Personnel radiation exposures and the release of radioactive effluents shall be maintained in accordance with the ALARA principle.

- No individual shall receive a radiation dose in excess of any regulatory limit.

Responsibility for establishing and ensuring adherence to these principles rests with the Senior Vice President, Field Operations. The General Manager has the overall responsibility and authority for the ALARA Program. The RPM is responsible for establishing and implementing the ALARA Program in accordance with written policies and procedures.

4.2.1 As Low As Reasonably Achievable Committee

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

The ALARA Committee:

- Communicates management's commitment to the ALARA Program;
- Monitors the implementation of the ALARA Program and serves as the advisor to ACP 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 ACP management and/or the Plant Safety Review Committee as 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 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 Production Support Manager, Operations Manager, Regulatory Manager, and an operations technician and/or a 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. The committee issues special reports prepared upon request of ACP management, or as determined by the chairperson.

The ALARA Committee reviews the ALARA Program and the review includes an evaluation of the results of audits performed by Health Physics (HP), reports of radiation levels, contamination levels, employee exposures, and effluent releases. 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 ACP management and the committee members. Recommendations of the ALARA Committee are documented and tracked to completion in the Corrective Action Program.

4.3 Organization and Personnel Qualifications

The RPM is responsible for providing guidance and direction for establishment and implementation of the RP Program and has direct access to the General Manager and Senior Vice President, Field Operations for radiological control matters. The RPM reports to the Production Support Manager, which provides independence from operations. The RPM and designee are required to have the technical competence and experience to establish RP programs (RPM qualifications are stated in Chapter 2.0 of this license application) and the management capability to direct the implementation and maintenance of RP programs.

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

HP Technicians perform the functions of assisting and guiding workers in the radiological aspects of the job. HP Technicians 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 plant or operational changes, including those initiated by changes to the Integrated Safety Analysis (ISA) for commercial ACP operations and for the HALEU Demonstration. These procedures are prepared, maintained and made available to appropriate personnel at the plant 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 the 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 areas where radioactive material may be encountered and by requiring that each person who enters those areas or facilities receive the appropriate level of radiological worker training. Personnel are trained commensurate with the hazard per 10 CFR Parts 19 and 20. Details concerning Visitor Site Access Orientation and radiological training are provided in Section 11.3.1 of this license application. The Radiological Worker Training Program addresses the requirements of 10 CFR 19.11 and 19.12 and workers' responsibilities under the Radiation Protection Program. The Radiation Worker Training program is described in Section 11.3.1.3 of this license application.

4.5.1 Visitor Site Access Orientation

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

4.5.2 General Employee Radiological Training

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

4.5.3 Radiation Worker Training

If a person requires unescorted access to the Restricted Area, radiological worker qualification is required. Radiation Worker Training is a biennial training requirement. Qualified Radiation Workers may be task qualified to perform selected HP Technician duties as authorized by the RPM.

4.5.4 Health Physics Technician

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

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

4.6 Ventilation and Respiratory Protection Programs

ACP building ventilation systems are described in Chapter 1.0 of this license application and in the ISA Summary. These systems are primarily designed to maintain the building environment required for proper operation of process and associated equipment.

There are no items relied on for safety (IROFS) identified with ventilation systems in the commercial ACP ISA Summary or its Addendum for the HALEU Demonstration. However, building ventilation systems are credited as defense in depth design features that help reduce the consequences of a UF_6 release in multiple analyzed events.

The ISA accident scenarios also identify use of portable ventilation units (commonly referred to as "gulpers") during applications ranging from pigtail operations to small-scale maintenance tasks to reduce worker exposure. In addition, administrative guidance requires the shutdown of building ventilation systems following detection of a UF_6 release to minimize the consequences to personnel (on and off site) during loss of confinement events.

4.6.1 Ventilation

In addition to general ventilation systems, and gulpers, portable ventilation units may be employed in the commercial ACP operation for short duration jobs when the unprotected worker could potentially exceed 0.8 Derived Air Concentration (DAC)-hours of exposure. These portable ventilation units are equipped with high efficiency particulate air (HEPA) filters and are designed to discharge room air at low velocities. (Gulpers are discussed in detail in Addendum 1 of the ISA Summary for the American Centrifuge Plant - HALEU Demonstration Section 3.8.2.)

The differential pressure of portable HEPA filtered ventilation units is checked per operating procedure for radiological purposes. 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 ACP. Portable HEPA filter unit use is normally specified on the RWP.

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

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

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

4.6.2 Respiratory Protection

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

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

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

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

Table 4.6-1 Contamination Levels

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

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

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

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

4.7 Radiation Surveys and Monitoring Program

The Radiation Surveys and Monitoring Programs are based on the requirements of 10 CFR Part 20, Subpart F 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 the monitoring program or results that exceed the administrative control levels are dispositioned in accordance with 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 surveys (refer to Section 4.8.2.4 below). Surveys are conducted to support plant 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 personnel. Decontamination is performed as appropriate considering the gained benefit from waste minimization, ALARA principles and worker access.

The routine survey program involves surveys 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 plant 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 ACP 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 Contamination Area (CA) or High Contamination Area (HCA) and actions will be 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

If required, to comply with the personnel monitoring requirements of 10 CFR 20.1502(a) and (b), 10 CFR 20.1202, and the reporting requirements of 10 CFR 19.13, 20.2106, and 20.2206, the ACP 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 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) situated in the ACP areas requiring a Criticality Accident Alarm System. A NVLAP accredited dosimeter reader processes dosimeters in the FNADs. The ACP maintains onsite capability to determine neutron flux and energy. 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 (plus or minus two weeks) unless authorized in writing by the RPM. The dosimeters may be supplemented, as appropriate, by radiation measurements made with radiation survey instruments.

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. The evaluation is performed to determine the types of activities that may have contributed to the worker's exposure. This may include, but is not limited to, procedural reviews, and review of work practices, work locations, and job assignments. Depending upon the conclusions of the evaluation, the individual may be allowed to continue radiological work; however, work restrictions may be imposed on individuals whose exposure exceeds the ACL.

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

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

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

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

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

4.7.4 Internal

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

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

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

The Licensee collects "random single void" urine samples from personnel. Isotopic analysis of 24-hour urine sampling are not routinely performed, however, 24-hour samples will be considered when dose assessments exceed 0.5 rem CEDE. Bioassay results are used to assign internal dose.

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

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

4.7.5 Airborne Radioactivity

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

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

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

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 in accordance with a procedure.

Table 4.7-1 Routine Contamination Survey Frequencies

Area Surveyed	Survey Frequency
Uranium Centrifuge Area	Yearly[a]
Contaminated Maintenance Areas	Quarterly
Contamination Control Zones (CCZ)	Quarterly
Lunchrooms/Breakrooms	Note c
Permanent Boundary Control Stations (BCS)[b]	Weekly
Change Rooms	Monthly
UF_6 Sample Handling Areas and Feed and Withdrawal Areas	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 working days (i.e., Monday through Friday). Weekends and plant 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
Dose Assessment Software	INDOS (Routine Analysis)

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]

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 exceptions to the applicable 10 CFR Part 20 requirements have been taken and require an exemption:

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

- Containers located in Restricted Areas within the ACP are exempt from container labeling requirements of 10 CFR 20.1904, as it is deemed impractical to label each and every container. In such areas, one sign stating that every container may contain radioactive material will be posted. By procedure, when containers are to be removed from contaminated or potentially contaminated areas, a survey is performed to ensure that contamination is not spread around the reservation.

- In lieu of the requirements of 10 CFR 20.1601(a), each High Radiation Area with radiation reading greater than 0.1 REM/hour at 30 cm but less than 1 REM/hour at 30 cm is conspicuously posted "Caution, High Radiation Area" and entrance into the area is controlled by an RWP. Physical and administrative controls to prevent inadvertent or unauthorized access to High and Very High Radiation Area is maintained.

4.8.2 Contamination Control

4.8.2.1 Access to Restricted Areas

Restricted Areas are areas to which access is limited 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 areas where radioactive material may be encountered and by requiring that each person who enters those areas 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 ACP 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. Personnel monitoring requirements are specified on RWPs. Equipment and materials are monitored and decontaminated if required prior to removal from CCZs and CAs, or are contained and controlled as radioactive material.

4.8.2.3 Personal Protective Equipment

Personal Protective Equipment (PPE) is provided for personnel entering contaminated areas. The type(s) of PPE required is consistent with the individual's work assignment and is dependent upon the type and level of contamination anticipated. Except for emergency evacuations, protective clothing is removed prior to exiting the Boundary Control Station 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 surface 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.

Total contamination in bulk, aggregate materials, or waste to be released for unrestricted use or disposal is specified in plant procedures.

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 100 microcuries (μCi) of beta and/or gamma emitting material or more than 10 μCi of alpha emitting material, other than ^{3}H, with a half-life greater than 30 days and in any form other than gas, 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.

Sealed plutonium alpha sources containing 0.1 μCi or more of plutonium, when not in use, are stored in a closed container designed and constructed to contain plutonium that might otherwise be released during storage. When in use, the ACP 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 ACP 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 ACP will file a report with the NRC 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. 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 ACP.

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

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

- Portable radiation detection and measurement instruments are inspected, maintained, and calibrated at least annually or removed from service.

- Instruments are calibrated following any maintenance, modification, or repair deemed likely to affect operation before being returned to service.

- Calibration sources and equipment used for dose rate instruments are within 5 percent (at 2 sigma) of the stated value and have documented traceability links to the NIST. Large area uranium slab sources are certified to 10 percent by NIST. Calibration sources used to calibrate contamination-monitoring equipment are within 20 percent (at 2 sigma) for activity and 10 percent (at 2 sigma) for surface emission rate.

- Portable HP instruments that are in use but do not have a built in automatic functional test feature are source checked daily, or prior to using the instrument if not used on a daily basis. Instruments with the automatic functional test feature that are in use are checked once a week.

4.8.5 Records and Reports

Radiological protection records demonstrate the effectiveness of the overall program and document personnel exposure. Records are maintained in the form required by 10 CFR 20.2110 and are retained as required by 10 CFR 20.2101 through 20.2106 according to the Records Management Program as outlined in Section 11.7 of this license application. 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 pursuant to 10 CFR Part 20, Subpart M; 10 CFR 30.50; 10 CFR 40.60; 10 CFR 70.50; and/or 10 CFR 70.74. Events requiring reporting to the NRC are investigated, tracked in a database, and monitored through completion in accordance with the Corrective Action Program. Details of reporting and notification for ACP incidents are described in Section 11.6 of this license application.

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 Posting Criteria (continued)

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 plant 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
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[a]
Ludlum 12 with alpha scintillator	Ludlum	Beta-gamma personnel contamination monitoring and removable contamination surveys	100 cpm above background[a]
RO20	Ludlum	Beta-gamma Dose/Dose rate	0 mR/hr - 5 R/hr

[a] 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.

[b] 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 Fuel Cycle Facilities License Applications*, Revision 2

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

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

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

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5.0 NUCLEAR CRITICALITY SAFETY

The American Centrifuge Plant (ACP) enriches uranium hexafluoride (UF_6). The commercial ACP operation is designed to enrich and safely handle up to 10 weight (wt.) % uranium-235 (^{235}U). The HALEU Demonstration Program is designed to enrich and safely handle uranium with an operational limit less than 20.0 wt. percent ^{235}U; however, enrichment levels up to 25 wt. % ^{235}U are authorized to permit for process fluctuations which can result in higher weight percent material. The maximum acceptable enrichment is identified for each operation evaluated for nuclear criticality safety (NCS). The specific authorized uses for each class of U. S. Nuclear Regulatory Commission (NRC)-regulated material are shown in Table 1.2-3 (commercial ACP operation) and Table 1.2-4 (HALEU Demonstration Program). 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, 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, these requirements are implemented through the ACP NCS Program.

In accordance with the requirements contained in 10 CFR 70.62, the likelihood and risks of an inadvertent nuclear criticality are evaluated in the Integrated Safety Analysis (ISA). The evaluation considers accident sequences caused by process deviations or other events internal to the facility and credible external events, including natural phenomena. Criticality Events are derived and evaluated through the process of generating Nuclear Criticality Safety Evaluations (NCSEs). In the case of the commercial ACP operation, Nuclear Criticality Safety Reports (NCSRs) were generated that will be transitioned to NCSEs prior to commencement of commercial plant operations. NCSEs will be developed based on the detailed design of the commercial ACP operation. If changes to the NCSEs or NCSRs are identified, the Licensee will revise the ISA, as necessary, to include any new or updated event sequence information, identify additional double contingency controls, or credit additional items relied on for safety (IROFS). The ISA includes credible nuclear criticality accident scenarios to assure that all nuclear processes are subcritical under normal and credible abnormal conditions. Additionally, preventative controls and measures are the primary means of protection against criticality in compliance with the performance requirements of 10 CFR 70.61(d).

The plant has established a threshold of 1 wt. percent or higher enriched ^{235}U and 100 grams (g) or more of ^{235}U for determining when an evaluation for NCS considerations of planned operations must be performed. This 100 g ^{235}U mass is a factor of 7 below the minimum critical mass, regardless of whether the material is optimally moderated and fully reflected. Based on this, the value is sufficiently low to use as a threshold limit. In view of this threshold, many of the ACP NCS Program features described in this chapter may not be required to be implemented for operations below the threshold. As described herein, the NCS Program provides the framework for a defense-in-depth philosophy to help ensure the risk of inadvertent criticality is maintained acceptably low. The NCS Program also provides the framework and resources for evaluating plant performance in establishing NCS analyses and controls for the design and operation of a uranium enrichment plant.

5.1 Management of the Nuclear Criticality Safety Program

5.1.1 Program Elements

The NCS Program described in this chapter is implemented by plant procedures. The NCS procedures address plant personnel NCS responsibilities, adherence to NCSE requirements, review and approval of fissile material operations, posting and labeling requirements, response to NCSE violations, and NCS training requirements. Controls and/or barriers that are relied on to prevent inadvertent criticalities are designated as IROFS in the ISA. The NCS Program meets the Baseline Design Criteria (BDC) requirements in 10 CFR 70.64(a)(9) concerning application of the double contingency principle in determining NCS controls and IROFS in the design of new facilities and new processes.

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 nuclear processes remain subcritical, and maintain an approved margin of subcriticality for safety;
- Establishing and maintaining NCS IROFS, based on current NCS evaluations;
- Providing training in emergency procedures in response to an inadvertent nuclear criticality;
- Complying with NCS BDC requirements in 10 CFR 70.64(a)(9);
- Complying with the NCS ISA Summary requirements in 10 CFR 70.65(b);
- Complying with the NCS ISA Summary change process requirements in 10 CFR 70.72; and
- Complying with the reporting requirements of 10 CFR 70.52 and 10 CFR 70 Appendix A.

5.2 Organization and Administration

5.2.1 Nuclear Criticality Safety Responsibilities

The ACP organization and administration are described in Chapter 2.0 of this license application. The General Manager assigns responsibilities and delegates commensurate authority to ACP 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 Director, Nuclear Safety is responsible for implementing the ACP NCS Program. The management reporting structure for the ACP is depicted in Chapter 2.0 of this license application. The Director, Nuclear Safety has direct access to the General Manager for nuclear safety matters and reports directly to the Senior Vice President, Field Operations.

The ACP 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 (hereafter referred to as fissile material operations) are identified and evaluated for NCS considerations prior to initiation of the operation. The organization managers or their designees are also responsible for ensuring NCS evaluations are requested, and for ensuring implementation of the requirements contained in the evaluations for these same operations. For those fissile material operations performed by personnel from multiple organizations, the General Manager assigns responsibility for that operation to a single organization manager or designee.

Management is responsible, in their respective operations, for ensuring that personnel are made aware of the requirements and limitations established by approved NCSEs 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 fissile material operations that do not have approved NCSEs 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 ACP NCS procedures will handle fissile material.

Managers/supervisors who are responsible for one or more fissile material operations are trained in NCS and ensure appropriate personnel receive NCS training as specified in ACP NCS procedures. This training provides personnel with the knowledge necessary to fulfill their NCS responsibilities. Section 11.3.1.4 of this license application discusses the NCS training program for those who manage, work in, or work near facilities where the potential exists for a criticality accident to occur (i.e. where fissile material handling/operations are performed).

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 Director, Nuclear Safety has, as a minimum, a bachelor's degree in engineering, mathematics or related science or equivalent technical experience, and six years nuclear experience, including six months at a uranium processing plant where nuclear criticality safety was practiced. The Director, Nuclear Safety 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 are prepared and technically reviewed by qualified NCS engineers. The NCS organization is independent of organizations that require NCSEs.

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

- Providing NCSEs 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 Plant Safety Review Committee (PSRC);
- Participating in the review of procedures that involve fissile material operations to ensure NCSE commitments have been effectively incorporated into operating procedures; and
- Participating in the review of work packages that involve fissile material operations, as requested.
- NCS group 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 by a reviewer with knowledge of the process and the NCS Program;
- 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

The NCS Program includes training and qualifications for NCS Engineers and Senior NCS Engineers which is based on the industry best practices provided in American National Standards Institute (ANSI)/American Nuclear Society (ANS) ANSI/ANS-8.26-2007, *Criticality Safety*

Engineer Training and Qualification Program.

The minimum requirements for a qualified NCS Engineer are:

- Bachelor's degree in engineering, mathematics, or related science;
- Familiarization with NCS by having a minimum of one year experience at a facility that process fissionable material where nuclear criticality safety was practiced;
- 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; and
- Performance of walk-through inspections under the guidance of a qualified NCS Engineer.

The Director, Nuclear Safety can modify the minimum qualified NCS Engineer qualification requirements for personnel who have worked for a minimum of three years at other facilities as an 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
- Approval by the Director, Nuclear Safety.

The Director, Nuclear Safety 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 These procedures contain the appropriate NCS controls for processing, storing, and handling fissile material. The NCSE requirements that specify employee actions are incorporated into procedures and are identified. Identifying these requirements ensures changes to these requirements are not made without review and approval by NCS. The NCSE requirements are incorporated into the appropriate procedures as required by the NCS Program 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 NCSE 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

Administrative NCS limits and controls for areas, equipment, and containers are presented through the use of postings and labels. Postings and labels are proposed, reviewed, and approved during the NCSE implementation process. The limits and controls are posted on the NCS requirements signs that are controlled and maintained according to the plant NCS procedures. Labels are also prepared in accordance with the plant NCS procedures and used as determined during NCSE implementation. Limits and controls are printed or written in an appropriate size, and the postings and labels are placed in conspicuous locations such that they are legible to the operator at the work location, on the specific component, item, or piece of equipment, or posted at the entrance to an operating area or storage area. The specific locations may be specified in the applicable NCSE or determined by the supervision responsible for the material.

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 plant's physical configuration. NCS controls that are IROFS are controlled as QL-2 items and NCS controls that are not IROFS are controlled as QL-3 items. The methodology for designating NCS engineered and administrative controls as IROFS is described in Section 3.1.2.3.2.7 for commercial ACP operations and Section 3.1.2.3.2.8 for HALEU Demonstration. 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 NCS engineered controls are described in NCSEs and the ISA. When an NCS engineered control is classified as an IROFS, the management measures described in the CM program associated with the QL-2 classification are applied. Some NCS controls associated with the commercial ACP operations are not IROFS and are classified as QL-3 items.

QL-3 is a quality grouping for structures, systems, and components required to fulfill the functions and meet the requirements established by the license application. For NCS controls that rely on certain structures, systems, or components, the portions of the CM program within the QL-3 classification as described in this section, as well as the following minimum features, are applied to those structures, systems, and components:

- Components are identified and controlled;

- Modifications are documented and reviewed;
- Change control process is applicable;
- Setpoints and tolerances are established for applicable components;
- Engineering drawings or specifications are provided;
- Procurement controls are provided; and
- Receipt inspection is used when specified.

Components and features that are identified in the NCSEs or the ISA are analyzed to determine the "boundary" of the system, encompassing those interconnecting and/or supporting items that are essential to ensure availability and reliability. The boundaries are identified on system drawings and/or other design outputs, and the configuration is verified to be as-built. These components and features are maintained in a design control document for the building or process. Each time a change is planned, the document is reviewed by the individual (e.g., design authority, system engineer, operations manager, maintenance, etc.) planning the change to determine if the change affects an IROFS/NCS control. Changes that could establish new fissile material operations or affect established fissile material operations are reviewed by NCS. The NCS Program establishes and maintains NCS safety limits for IROFS/NCS controls and maintains adequate management measures to ensure the availability and reliability of the IROFS/NCS controls. Operating limits may be established during flow down of NCS safety limits to ensure their continued reliability and availability.

The change control process specifies the organizations required to perform reviews of changes. Changes that affect existing fissile material operations are evaluated by NCS to determine 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 and placing the affected structures, systems, or components in service.

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. 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. These activities are specified in ACP procedures.

Operating fissile material process areas are reviewed on a regular basis through a combination of walk-throughs and reviews by work crew supervision. 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 ACP NCS procedures. 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, fissile material operating areas are assessed by NCS personnel via walk-through at least annually, sometimes in conjunction with the assessments discussed below.

Work crew supervision provides real-time assessments of fissile material operations within their operating area to ensure NCS requirements are being adequately implemented and operating conditions have not been altered to adversely affect NCS. Fissile material operations management also performs an annual self-assessment to ensure NCS program requirements are being met in the field.

In addition to the annual self-assessments, independent internal audits of the NCS Program are conducted or coordinated by the Piketon Quality Assurance Manager as described in Section 11.5 of this license application. The purpose of these audits is to determine the adequacy of the overall NCS Program. This includes the adequacy of the NCSEs, 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. If a condition is identified that is non-compliant with NCS program requirements, field personnel are to report the condition as directed by plant procedures. If the condition is not covered by an existing procedure, consultation with a qualified NCS engineer is required before taking any corrective action. Immediate corrective actions may be provided by the responding NCS engineer verbally or in writing. NCS emergency response is discussed in Section 5.4.2 below and is described in more detail in Chapter 8.0 of this license application.

Managers in charge of fissile material operations are provided additional training on NCS and response to NCS deficiencies as described in Section 11.3.1.4 of this license application and the ACP NCS procedures. Each NCS non-compliance is evaluated by an NCS engineer to determine the impact on double contingency and 10 CFR 70.61 performance requirements. The availability and reliability of credited controls from the applicable NCSE are ascertained to support the shift supervisor in determining safety significance and reporting requirements. The evaluation for reportability of events is based on whether the controls were lost or degraded (i.e., whether they were unreliable or unavailable to perform their safety functions) resulting in the failure to meet 10 CFR 70.61 performance requirements, not based on whether the safety limit of the associated parameter was actually exceeded.

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. One-hour reportable events are significant operational events that must be quickly reported. Such events typically do not require substantial evaluation to determine reportability (e.g., an event involving the loss of all controls, such that a criticality accident is possible). If it cannot be determined whether an NCS incident requires reporting under paragraph (a) of 10 CFR Part 70, Appendix A, the NCS incident should be reported within one-hour of discovery. Twenty-four-hour reportable events have less safety significance than one-hour reportable events and sometimes require more extensive evaluation to determine reportability. The twenty-four-hour time period for reportable events is intended to allow sufficient time to make this determination. If the determination cannot be completed within this time frame, then the NCS incident is reported within twenty-four hours of discovery. The time of discovery begins when a cognizant individual observes, identifies, or is notified of the NCS safety significant event or condition. A cognizant individual is an individual who, by position or experience, is expected to understand that the condition or event adversely impacts double contingency and 10 CFR 70.61 performance requirements.

The deficiency data is trended to monitor and prevent future violations. Corrective actions are taken for identified deficiencies in accordance with the Quality Assurance Program Description for the American Centrifuge Plant and the Corrective Action Program as described in Section 11.6 of this license application. 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 requirements of American National Standards Institute (ANSI)/American Nuclear Society (ANS) ANSI/ANS-8.1-2014, ANSI/ANS-8.3-1997, ANSI/ANS-8.19-2014, ANSI/ANS-8.20-1991, ANSI/ANS-8.21-1995, ANSI/ANS-8.23-2007, ANSI/ANS-8.24-2017, and ANSI/ANS-8.26-2007 standards as discussed in this section with the exceptions noted in Section 1.4.

5.4.2 Nuclear Criticality Safety Evaluation

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 evaluation describes the scope of the operation, evaluates credible criticality accident contingencies, and establishes NCS requirements to maintain the operation subcritical. The evaluation process is governed by written procedures.

When an NCSE (or a change to an existing NCSE) is needed for a particular fissile material operation, a request is submitted to the NCS group to evaluate the proposed operation. Other methods for initiating an NCS change include, but are not limited to: 1) the engineering change process, and 2) the corrective actions process, self-assessments, and external audits and inspections.

In response to the request, an NCS evaluation may be performed or the request may be returned due to inadequate detail, the change is bounded by a current analysis, or the operation does not involve uranium enriched to 1 wt. percent or higher ^{235}U and with mass of 100 g or more ^{235}U (see Section 5.4.2.1). If necessary, a NCSE is prepared (or an existing NCSE is revised) to document the analyses performed as specified in the NCS evaluation 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.

Fissile material operations must comply with the double contingency principle. The double contingency principle as stated in ANSI/ANS-8.1-2014, Section 4.2.2, is "Process designs should 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 phrase "changes in process conditions" does not imply that reliance on two different parameters is required to satisfy the double contingency principle. The double contingency principle is satisfied by implementing the controls necessary to ensure at least two unlikely, independent, and concurrent changes in process conditions would have to occur before a criticality is possible. Process conditions include the characteristics of a process that have an effect on nuclear criticality safety, such as parameters, environment, and operations. Controls are applied as necessary to ensure each change in process conditions is unlikely to occur.

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 uranyl fluoride 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. Application of the double contingency 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 NCSE will document the basis for the conclusion that a change in a process condition is "unlikely." 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. Where practical, the use of explicit NCS controls will be used as the preferred approach over the reliance on natural and credible course of events. The parameters or conditions relied on and the limits must be specified and justified in the NCSE. Reliance on two different parameters is preferred over reliance on multiple controls on a single parameter. If relying on two or more controls on a single parameter, diverse (i.e., different means of controlling the parameter) is preferred over redundant means of control. Management measures described in Chapter 11.0 of this license application and other safety programs are sometimes used to help ensure a change in a process or parameter is "unlikely." For example, the Radiation Safety Program and/or the Fundamental Nuclear Material Control Plan may be credited with providing controls on fissile material handling; the Fire Safety Program may be credited with providing controls on combustible material loading and/or hot work activities in fissile material processing/storage areas; the Procedures Program may be credited with ensuring compliance with procedures; etc.

Where the natural or credible course of events is relied upon in whole or in part to prevent a process condition change, no specific additional controls will be necessary to maintain them. The factors that influence the process are 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 plant changes could adversely affect the intended criticality safety function of the items relied upon for criticality safety), specific NCS controls are 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 may be used in the evaluation process to examine potential accident scenarios. Qualitative or quantitative estimates of event frequency may be 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 technical review of an NCS evaluation is performed by a Senior NCS Engineer or an NCS Engineer completing the technical review under the guidance of a Senior NCS Engineer.

The NCSE documents the NCS requirements for the operation. The NCS requirements 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 NCSE approval process involves the acceptance of the NCSE by the technical reviewer. The supervisor of the affected operation also reviews the NCSE to confirm the NCSE adequately identifies normal and credible abnormal conditions and establishes requirements that are verifiable and compatible with the planned operation. The Director, Nuclear Safety performs a review to ensure consistency with other NCSEs and other potentially conflicting requirements or regulations.

After approval by the Director, Nuclear Safety, a review is 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 NCSE is required. PSRC approval is required for initial NCSE approval and for changes that impact the ISA Summary. Editorial changes require only the approval of the Director, Nuclear Safety. Editorial changes are defined as changes that do not change the technical basis of the NCSE. Once approved, the NCS controls, limits, evaluation assumptions, and safety items are verified to be fully implemented in the field. The operating 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 NCSE. Operational aids such as postings, labels, boundaries for fissile material operations, and fissile material movement guidelines may be used to implement the NCSE. The manager/supervisor ensures postings and labels are prepared and verify that they are properly installed to support implementation of the NCSE. The procedures are prepared or modified to incorporate the NCSE requirements. Managers/supervisors are responsible for ensuring the employees understand the procedures and understand the NCS requirements before the work begins.

Each completed NCSE is issued as a controlled document. Completed NCSEs are archived and retrievable as permanent quality records in accordance with the RMDC requirements described in Section 11.7 of this license application. The 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.

5.4.2.1 Non-Fissile Material Operations

Some operations involve situations in which the uranium has an enrichment of less than 1 wt. percent ^{235}U or an inventory of less than 100 g ^{235}U. These operations are termed "non-fissile material operations" and are performed without the need for NCS double contingency controls. The determination of which operations are fissile versus which operations are non-fissile are made by NCS and may be contained within a NCSE or as a separate document. The determination of an operation being non-fissile must include normal and credible abnormal upset conditions to ensure the enrichment and/or inventory are maintained below 1 wt. percent ^{235}U or below 100 g ^{235}U. Controls are sometimes applied to a non-fissile material operation to ensure it does not inadvertently involve fissile material. This determination is made by an NCS engineer in collaboration with the responsible line manager.

5.4.3 Design Philosophy and Review

Through the CM Program, designs of new fissile material equipment and processes must be approved by NCS before implementation. Where practical, the use of engineered controls on mass, geometry, moderation, volume, concentration, interaction, or neutron absorption will be used 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 if they may credibly degrade such that control of the parameter is jeopardized.

The preferred design approach establishes a hierarchy of controls. The use of passive engineered controls; in particular, passive engineered geometry control is the most preferred. The order of preference for NCS controls is (1) passive engineered, (2) active engineered, (3) enhanced administrative, and (4) simple administrative controls. The adherence to the preferred design approach is utilized 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. Deviations from the preferred design approach are justified in supporting documentation to the NCSEs.

Fissile material equipment designs and modifications are reviewed to ensure that engineered controls are used for NCS to the extent practical. Administrative limits and controls will be implemented to satisfy the double contingency principle for those cases where the preferred design approach is not practical.

5.4.4 Criticality Accident Alarm System Coverage

A criticality accident alarm system (CAAS) that complies with 10 CFR 70.24 and ANSI/ANS-8.3-1997 is provided to alert personnel if a criticality accident occurs. The system utilizes an audible and/or visual signal to alert personnel in the area to evacuate to reduce radiation exposure resulting from the incident.

The need for CAAS coverage is considered during the development process for NCS evaluations. In general, coverage is provided for fissile material operations, except the UF_6 cylinder storage yards as specified in Section 1.2.5 of this license application. Other exceptions to CAAS coverage are documented in NCS evaluations and are based on a conclusion in the NCSE that a criticality accident is non-credible in the area where the fissile material operation is ongoing. Conclusions of non-credibility require at a minimum that the inventory of ^{235}U in the area is less than 700 g. In addition, CAAS is not required for areas having material that is either packaged or stored in accordance with 10 CFR Part 71 or specifically exempt according to 10 CFR 71.15. Areas that do not contain fissile material operations do not require a NCSE and do not require CAAS coverage.

The CAAS is designed to detect gamma radiation levels that would result from the minimum criticality accident of concern as defined by ANSI/ANS 8.3-1997 and to provide annunciation by audible evacuation alarms that are supplemented by visual alarms in some areas, such as high-noise areas. A secondary function is to activate the building radiation warning lights and alarms at the X-3012 Process Support Building Area Control Room (ACR).

For each area requiring CAAS coverage, a monitoring system is installed that provides coverage of the area by at least one detection unit. A detection unit is a set of at least three radiation detectors that may be co-located or may be distributed over the area. The detection logic of the system requires that two of the three detectors must be activated to initiate the building evacuation alarm system. Each detector may be logically part of more than one detection unit.

The building evacuation alarm system includes interior CAAS evacuation horns and radiation warning lights to deter personnel from entering the area after an evacuation. In addition, facilities within 125 feet of a fissile material operation area requiring CAAS coverage have radiation evacuation horns installed inside and radiation warning lights installed to prompt evacuation and deter personnel from entering the area. Personnel who have routine access to these facilities have been trained to recognize and respond to these indications as described in Section 11.3.1.1.2 of this license application.

To protect against the loss of coverage, the CAAS includes redundant decision logic, a backup power supply, detector status information and system self-diagnostic information are provided to the X-3012 building ACR. The CAAS has been designed to survive and/or withstand credible abnormal events as described in the accident analysis for a sufficient time to warn personnel to evacuate. In the event CAAS coverage is lost for an operation, plant procedures provide for compensatory actions, which may include shutdown of equipment, limiting access, halting movement of uranium-bearing material, or other actions, such as use of personal alarming dosimeters for personnel that must access the area during a CAAS outage.

Potential criticality accident locations and predicted accident characteristics are evaluated and documented in sufficient detail to assist in emergency planning as described in ANSI/ANS-8.23-2007. Additional information regarding nuclear accident planning and response is discussed in Chapter 8.

5.4.4.1 Portable CAAS

In the event a fissile material operation requiring CAAS coverage is performed beyond the detection range of established CAAS instrumentation, a portable unit may be used. The portable unit has the same detection capabilities as the permanently installed units. Alarm annunciation, however, is usually limited to the immediate area within the audible range of the unit's alarm with an additional telemetric link to the X-3012 ACR. This link will transmit the location of the unit, if mobile, and allow the use of the plant PA system to warn personnel within 125 feet of the area of the portable unit to evacuate. A portable unit will not be used for more than 24 continuous hours and it may be located indoors, outdoors, or on a vehicle.

If fissile material operations in an area without a permanently installed CAAS are required to exceed 24 continuous hours, all personnel not directly involved in the affected operations, or otherwise required for the safety or security of the facility, will be evacuated from an area within a 125 foot radius of the fissile material until the operations are concluded. In addition, affected operations shall be terminated as soon as safely achievable.

5.4.5 Technical Practices

5.4.5.1 Application of Parameters

Provided below are general criteria associated with application of nuclear parameters.

- Each parameter is assumed to be at its optimal or most reactive credible value unless specified controls are implemented to limit the parameter to a particular range of values.
- When process variables can affect the normal or most reactive credible values of parameters, controls to maintain the variables are established, and the basis for the correlation between the process variable and associated controlled parameter is documented.
- When instrumentation is relied on for measuring a parameter credited for NCS, instrumentation subject to facility management measures is used.
- When measurement of a single parameter is used as the sole basis for double contingency, independent means of measurement are used.
- Safety limits on controlled parameters are established and/or implemented with sufficient margin to account for tolerances and uncertainties.

The nuclear parameters which can impact nuclear criticality safety are summarized below, along with examples of how the parameters are controlled at the ACP. More detail on the technical practices associated with evaluating and implementing controlled parameters is provided in the NCS program procedures.

Moderation

Water is considered to be the most efficient moderator commonly found in the ACP. This is because optimally moderated UO_2F_2/water solutions are more reactive than the oils allowed in the centrifuge process gas equipment or equipment connected to the process gas system. (Reference 16). 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. Generally, moderation control is not maintained by measurement; however, when used, dual independent sampling methods are implemented.

Moderation control is applied to prevent moderators (other than moderation due to air in-leakage) from entering plant equipment containing UF_6. In areas where greater than the safe mass of uranium (as defined below) is handled, processed, or stored and moderation controls are applied, that facility's pre-fire plan (reference Section 7.1.4 of this license application) includes any unique firefighting strategy or tactics that may be needed to limit the use of moderator material. However, even in these areas, the application of the double contingency principle ensures the worst credible loss of moderation control cannot result in a critical configuration without an additional independent and concurrent upset event.

The centrifuge process equipment is comprised of a variety of closed systems designed to process gaseous UF_6. This closed system minimizes the introduction of moderation due to wet air in-leakage. Because UF_6 reacts chemically with moisture (a moderator) to produce solid uranium-bearing compounds that impedes the proper operation of the process equipment, the UF_6 bearing systems are designed to minimize introduction of moisture.

Moderating materials can be present as interstitial moderators that are in solution or intermixed into the fissionable material compound (e.g., water in uranyl fluoride solution). Moderating materials may also be present as interspersed moderators that exist as moderating materials located between distinct lumps or regions of fissionable material (e.g. sprinkler activation). Interspersed moderation issues are discussed in the *Reflection* section, below.

Volume

Volume limits are used as specified in NCSEs. 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. When volume control is used, the size of the containers or equipment is ensured through the CM Program and/or by procedurally requiring the use of certain containers for fissile material operations.

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 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. In accordance with the preferred design approach, described in Section 5.4.3 of this chapter, passive engineered controls are used to the extent possible to ensure spacing requirements are maintained. When used, the structural integrity of the spacers or racks is sufficient to maintain spacing for normal and credible abnormal upset conditions.

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 NCSEs, are maintained by the CM Program, and are verified prior to authorizing initial operation. "Safe geometry" is a term typically used to describe systems that are not dependent on any other nuclear parameter for criticality safety. "Favorable geometry" is a term typically used to describe systems that rely on one or more stated parameters to maintain criticality safety. However, the use of these terms is not rigidly applied throughout the available literature. Both "safe geometry" and "favorable geometry" dimensions may be obtained from established standards or operation specific reactivity calculations.

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. Safe mass values may be obtained from established standards or operation specific reactivity calculations. "Safe mass" is defined as the quantity of fissile material that is safely subcritical under the most reactive credible conditions (defined for a given isotopic composition and physiochemical form), including allowance for over-batching. Experimental data is not used as the sole source for safe mass values. Safe mass values are chosen to ensure no single credible upset can result in a critical configuration. When safe mass values are dependent on the geometry, enrichment, composition, or some other parameter, the combination of mass and the other parameter is used as one control to meet the double contingency principle.

The safe mass values are communicated to the operating personnel via the operating procedures. Unless specifically controlled, an item containing enriched uranium is assumed to contain the most ^{235}U credible based on the available volume. When mass is determined through measurement, instrumentation that is subject to management measures is used.

Enrichment

The maximum ^{235}U enrichment for each operation is established by the specific NCSE. Credible process upset conditions that could alter the ^{235}U enrichment are also considered in the NCSEs. When the enrichment of uranium needs to be measured for an NCS control, the measurement is obtained using either installed equipment or based on samples analyzed in a laboratory.

Uranium-containing material in the ACP 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 ^{235}U enrichment of UF_6 in the ACP HALEU cascade is limited to less than 20 wt. percent with the potential for momentary enrichment transients up to 25 wt. % ^{235}U during HALEU cascade operations. Small quantities of greater than 20 wt. percent ^{235}U may also be present outside of plant equipment in the form of standards.

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 NCSE for the operation and it will be measured using instrumentation.

UF_6 in the gaseous phase, at any credible pressures and temperatures existing in the plant equipment, is incapable of supporting a nuclear chain reaction even when intermixed with hydrogenous material (e.g., hydrogen fluoride [HF]). UF_6 in the gaseous phase in plant equipment has low material density.

Heterogeneity

Heterogeneous configurations are considered for those operations that involve small fissile material and moderator regions. Means of causing inhomogeneity are evaluated and controlled as needed depending on their effect on subcriticality. Assumptions that can affect the physical scale of heterogeneity are based on observed physical characteristics.

Concentration

Concentration controls are used on a case-by-case basis. When the criticality safety of an operation depends solely 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 NCSE for the operation involved. The specific controls and details are documented in the NCSE for each operation that relies on concentration controls. Precipitating agents, including freezing, are controlled as necessary to ensure they do not inadvertently affect solubility or homogeneity or increase the concentration.

Reflection

Normal and credible abnormal reflection is considered when performing NCS evaluations. The possibility of full water reflection is considered when performing analyses. Interspersed moderation is evaluated with either full water reflection or water films with a bounding water density value to simulate sprinkler activation or precipitation combined with full density water blocks to simulate personnel. It is recognized that concrete can be a more efficient reflector than water, and its potential presence is considered. If special moderators such as deuterium, beryllium, or graphite, or if large amounts of hydrogen-rich materials (e.g., hydrocarbon oil or polyethylene, etc.) are present, the NCS evaluation ensures the modeled reflection conditions remain bounding. Reflection controls are used to limit the potential reactivity of a fissile material operation.

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 NCSE and ANSI/ANS-8.21-1995, *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*. 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 NCSE. Soluble neutron absorbers are not credited by the ACP NCS Program.

An 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, neutron spectra, and uncertainties in the neutron cross-sections.

5.4.5.2 Methods of Calculation

Experimental Data

Experimental data are used for validation of the computer code used to perform the calculations needed to support the development of NCSEs. The experimental data used are discussed in the code validation report (Reference 15).

Handbooks and Standards

Handbooks and standards (e.g., ANSI/ANS-8.1-2014) are also used in some cases when simple systems are being evaluated. Handbooks and standards used for ACP operations are nationally recognized throughout the NCS industry as high quality analyses that have been confirmed through many years of use or based on experimental data. Most of the operations performed in the ACP are too complicated to be adequately addressed by data in a handbook/standard. When isolated operations are performed with small amounts of fissile material, referencing handbooks/standards is useful to support conclusions in the NCSE. Examples of the handbooks used include, but are not limited to, ARH-600, *Criticality Handbook* and LA-10860-MS, *Critical Dimensions of Systems Containing* ^{235}U, ^{239}Pu, *and* ^{233}U. Other handbooks are held to similar criteria for excellence, industry acceptance, and quality of data to be used at the ACP without further verification calculations.

Because handbooks and standards tend to give minimum critical or maximum subcritical values, use of these values for criticality controls is not appropriate to meet the double contingency principle. Instead, these values are reduced such that subcriticality can be demonstrated under normal and credible abnormal conditions.

Hand Calculations

Applicable methods for evaluating single units include Modified Two Group Diffusion Equation (i.e., Critical Equation), Buckling Conversion, and Comparative Analysis..

- **Modified Two Group Diffusion Equation** – This method is applicable to, and most widely used for, solution systems.
- **Buckling Conversion** – The method of buckling conversion or shape conversion is applicable to all materials.
- **Comparative Analysis** – This method involves direct comparison of the system configurations to subcritical data from NCS handbooks.

Applicable methods for evaluating arrays include the Solid Angle Method and the Surface Density Method using unit shape factor.

- **Solid Angle Method** – This method is applicable to solution systems. It is not useful if reflection is more effective than a thick water reflector located at the array boundary. The conditions that must be satisfied in order to successfully apply the solid angle method are (1) $k_{effective}$ (k_{eff}) of any unreflected unit does not exceed 0.80; (2) each unit is subcritical when completely reflected by water; (3) the minimum surface-to-surface separation between units is 0.3 meters; and (4) the allowed solid angle does not exceed 6 steradians.

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

When hand calculations are used, the specific methodology employed will be based on industry-accepted methods (e.g., areal density, solid angle technique, etc.), subject to the limitations of those methods.

Computer Calculations

NCS computational analyses which involve the calculation of k_{eff}, may be used to determine whether the system will be subcritical under both normal and credible abnormal process conditions. Computer codes that simulate the behavior of neutrons in a process system or that solve the Boltzmann transport equation are used.

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

See Appendix F of this license application for a description of the computer code validation applicable to the HALEU Demonstration Program. Within 30 days of making any non-administrative changes to the validation report, the Licensee shall provide the Commission with a summary of changes and shall provide the revised validation report upon request. The licensee may not implement changes to reduce the margin of subcriticality for safety (i.e., factors or methods that would adversely affect the Upper Subcritical Limit) without NRC approval of the change.

5.5 References

1. ANSI/ANS-8.1-2014 *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*

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

3. ANSI/ANS-8.19-2014, *Administrative Practices for Nuclear Criticality Safety*

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

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

6. ANSI/ANS-8.23-2007, *Nuclear Criticality Accident Emergency Planning and Response*

7. ANSI/ANS-8.24-2017, *Validation of Neutron Transport Methods for Nuclear Criticality Safety*

8. ANSI/ANS-8.26-2007, *Criticality Safety Engineer Training and Qualification Program*

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

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

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

12. NRC Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, Revision 3

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

14. NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications,* Revision 2

15. EE-3101-0013, *NCS Code Validation of SCALE 6.2.3 and Cross Section Set v7-252 for* k_{eff} *Calculations*, Rev. 0, December 2019

16. DAC-3101-0006, *Safe Mass Study for* UF_4 *and Oil*, February 2020

17. "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/DOC (95) (03), Nuclear Energy Agency Science Committee, Organization for Economic Co-Operation and Development, July 2018 Edition.

18. Jordan, W.C., Landers, N.F., Petrie, L.M., "Validation of KENO V.a Comparison with Critical Experiments," ORNL/CSD/TM-238, Martin Marietta Energy Systems, Contract Number DE-AC05-84OR21400, December 1986.

6.0 CHEMICAL PROCESS SAFETY

The American Centrifuge Plant (ACP) operations require limited quantities of radioactive, hazardous, and toxic chemicals for maintenance and production activities that are performed in support of the basic uranium enrichment process. For the ACP Commercial Plant, these chemicals are discussed in the Integrated Safety Analysis (ISA) Summary for the American Centrifuge Plant, Chapters 5.0 and 6.0, as well as their appendices. For the ACP HALEU Demonstration, these chemicals are discussed in Addendum 1 of the ISA Summary for the American Centrifuge Plant HALEU – Demonstration, Chapters 5.0 and 6.0, as well as their appendices. Pursuant to 10 *Code of Federal Regulations* (CFR) 70.62, the plant safety program includes process safety information to address hazardous materials.

This chapter summarizes the chemical process safety program for the ACP, the integration of chemical safety with uranium enrichment operations, and the management systems used by the plant for chemical safety. A description of the plant and uranium enrichment process is provided in Section 1.1 and a description of the reservation is provided in Section 1.3 of this license application. The uranium hexafluoride (UF_6) inventory that is integral to enrichment is addressed in the ISA Summary. The risks associated with UF_6 and its airborne release reaction products, hydrogen fluoride (HF) and uranyl fluoride (UO_2F_2), are discussed in the ISA Summary, Sections 5.2.1, 5.2.1.1, 5.2.1.2, 6.1.1, 6.1.1.1, 6.1.1.2, 6.1.1.3, and 6.1.1.4; and Appendix D, Sections D.1 through D.16 for the ACP Commercial Plant. The risks associated with UF_6 and its airborne release reaction products, HF and UO_2F_2, are discussed in Addendum 1 of the ISA Summary for the American Centrifuge Plant HALEU – Demonstration, Sections 5.1, 6.1.1.1, 6.1.1.2, 6.1.1.3, 6.1.1.4, 6.1.1.6, and 6.1.1.7; and Appendix D, for the HALEU Demonstration.

The ACP chemical process safety program 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 Radiation Protection Manager/Supervisor is responsible for the plant chemical process safety program. Chemical safety incorporates engineering and administrative controls to manage risk. Prevention is the preferred approach. Workers use personal protective equipment (PPE) when it is specified in procedures.

6.1 Process Chemical Risk and Accident Sequences

Chemical inventories at the ACP are maintained below the threshold quantities 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); therefore, these regulations do not apply to the ACP.

Chemical safety consists of the integration of environmental, safety, and health management systems to address chemical hazards. Chemical safety controls are designed to prevent 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 and Industrial Hygiene and Safety (IHS) programs are utilized.

Chemical safety controls are limited to non-radiological materials. Radiological materials are addressed throughout the ISA Summary for the ACP Commercial Plant and Addendum 1 of the ISA Summary for the American Centrifuge Plant HALEU – Demonstration and in Chapter 4.0 of this license application. Chemical process safety is addressed in the ISA. The ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration, Chapter 6.0 identifies potential accident sequences and Chapter 7.0 designates selected controls (i.e., items relied on for safety [IROFS]) to either prevent such accidents or mitigate their consequences to an acceptable level.

Chemicals with significant radiological impact are limited to UF_6 and its release products, HF and UO_2F_2, as indicated in Sections 5.1 and 5.2 of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration. Other chemical hazards, which are not considered to have any radiological impact, are listed in Appendix B of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration. Techniques and assumptions for estimating airborne concentrations and predicting toxic footprints from chemical releases are presented in Appendix D of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration, which also presents source terms and vapor dispersion models used to calculate airborne concentrations of UF_6 and its release products. The American Industrial Hygiene Association (AIHA) Emergency Response Planning Guidelines (ERPGs) have been selected as the chemical response standard for the ACP. The ERPGs provide airborne concentration limits to effectively protect individuals against toxic exposure to hazardous chemicals. These guidelines are discussed in Appendix A of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration.

Management measures are established to provide reasonable assurance of the availability and reliability of IROFS. The ISA includes consideration of the toxicity of uranium, radiological hazards, and chemical hazards that may impact radiological safety. The details of the analysis are provided in the ISA Summary.

6.2 Items Relied on for Safety and Management Measures

Safety in normal operations is maintained through implementation of the defense-in-depth engineering design philosophy. The ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration describe the basis for providing successive levels of protection such that health and safety of employees and the public is not wholly dependent upon any single element of the design, construction, maintenance or operation of the facility. The schemes employed to ensure safe operation of the ACP include management measures that provide for the reliability of IROFS. These measures include configuration management (CM), maintenance, procedures, training, surveillance, and testing. Management measures are described in Chapter 11.0 of this license application.

6.2.1 Items Relied on for Safety

Chemical process safety controls that prevent accidents or mitigate their consequences are identified in Section 7.2 of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration. These controls are designated as IROFS and address the chemical hazards that may impact radiological safety. Tables 6.1-1, 6.1-2, 6.1-3, and 6.1-4 of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant HALEU – Demonstration, identify both radiological and non-radiological accident sequences with regard to performance criteria. These are also discussed in Section 7.3 of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration.

6.2.2 Management Measures

Each of the management measures that helps 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

Centrus Energy Corp., with approval of the DOE, assigned the sublease for the space for the ACP (including the HALEU Demonstration) to the Licensee, ACO. The Licensee subleases, from Centrus Energy Corp., certain support buildings/facilities on the DOE reservation. The ACP and the DOE have their own chemical safety programs and share information regarding hazardous chemicals used by each entity. The DOE environmental restoration contractors and sub-contractors may also be present on the reservation. The DOE provides information regarding any hazardous chemicals used by these "third-parties" that could impact ACP operations. Third-party chemicals are covered by a shared site agreement with DOE and reviewed in accordance with procedures.

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

- Lockout/Tagout
- Hazard Communication
- Confined Space Entry
- Safety and Health Work Permit

- Hot Work Permit
- Personal Protective Equipment
- Signs/Labeling/Tagging
- Safety Training

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

6.2.2.2 Training

The Training and Procedures Manager has overall responsibility for employee training. ACP operators, 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 plant Training Program described in Section 11.3 of this license application and occurs at the job-specific level.

Contractor (typically construction, maintenance, and service) personnel receive access training and plant-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. The Site Technical Representative is the liaison between the contractor and the Licensee. If construction activities interface with chemical systems, ACP representatives ensure appropriate job review, training, and guidance is provided.

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) for the American Centrifuge Plant.

Engineering develops maintenance and inspection requirements and criteria for chemical systems in conjunction with the specific plant maintenance organization, manufacturer's recommendations, and ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration. These chemical safety requirements are based on the functions of IROFS identified in the ISA Summary, Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration, 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 plant maintenance programs. These plant 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. ACP personnel perform inspection and testing to fulfill requirements for quality in accordance with the CM Program, which is described in Section 11.1 of this license application.

Independent overview of maintenance activities on chemical system hardware and requirements are addressed by the QAPD and 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
- Plant Safety Review Committee (PSRC)

The pre-operational safety review process is conducted in accordance with program implementing procedures. The scope of the safety review is determined by the PSRC which 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. Director, Engineering, as the design authority for the ACP and HALEU Demonstration, administers the CM Program. The CM Program includes an organizational structure and administrative processes and controls to ensure that accurate, current design documentation is maintained that matches the building physical configuration.

6.2.2.5 Emergency Planning

Emergency Management is described in Chapter 8.0 of this license application. The Emergency Management Plan for the ACP outlines the roles and responsibilities of personnel during an emergency and describes the emergency response measures, including on-site and off-site protective actions.

Chapter 8.0 Section 8.1 details that No Emergency Plan as discussed under 10 Code of Federal Regulations (CFR) 70.22(i) is needed for the HALEU Demonstration Program.

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 plant "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 plant procedures direct an employee response to a minor spill, an employee can implement the plant response procedure after "See and Flee" requirements have been accomplished and the area may be reentered.

6.2.2.6 Incident Investigation

Identification, reporting, and incident investigation, described in Section 11.6 of this license application, are conducted in accordance with plant procedures. The level of investigation is based upon severity and significance of the event, as well as the regulatory requirements involved. Unacceptable performance deficiencies are addressed in accordance with the ACP Corrective Action Program. Documentation is retained in accordance with RMDC requirements described in Section 11.7 of this license application.

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

6.2.2.7 Audits and Inspections

Formal audit responsibilities are assigned to the Piketon Quality Assurance Manager. In addition, internal organizations have monitoring programs, assessments, and reviews as required by program implementation procedures. The 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 programmatic requirements that apply to Quality Level (QL)-1 and QL-2 items. These quality assurance elements and requirements apply to chemical safety items classified as QL-1 or QL-2 in a graded approach, as described in the QAPD. Additional discussion regarding the ACP graded approach to quality assurance is provided in Chapter 11.0 of the License Application.

6.2.2.9 Human Factors

Human factors design responsibility for plant and system design in the ACP 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. The Human Factors Engineering program is described in Section 2.6 of the ISA Summary.

6.2.2.10 Detection and Monitoring

Chemicals with significant radiological impact such as UF_6, HF, and UO_2F_2 that are processed in the various ACP facilities are provided with detection and monitoring systems to identify chemical releases as appropriate to the release event. Non-radiological chemicals that do not have significant radiological impact are maintained below PSM/RMP threshold quantities and do not require detection and monitoring.

6.2.2.11 Chemical Safety Control Strategy

The 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, Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration, 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/controlled and to ensure that inventories are maintained below PSM/RMP threshold quantities. Material Safety Data Sheets/Safety Data Sheets (MSDSs/SDSs) are maintained in a central location in the ACP and are available at all times to plant employees, including emergency response and fire department personnel from on- and off-site. The first process identifies and inventories chemicals used at the ACP. This process ensures that chemicals used at the plant are appropriately addressed for safety. The process includes:

- Purchase requisition reviews;
- A listing of chemicals used;
- A centrally-located MSDS/SDS library, which is maintained and routinely updated by Industrial Hygiene; 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 changes to hazardous chemical inventories or locations exist as a result of a request for a new, modified, or decommissioned building, process or storage location, an appropriate chemical safety review is applied to address regulatory requirements. Physical changes to the plant, 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 the latest revision of MSDSs/SDSs for these chemicals. Hard copies are maintained by the contractor at the job site, by Industrial Hygiene in a central location, and by appropriate Facility Custodians.

6.2.2.11.2 Chemicals Addressed By Integrated Safety Analysis Summary

The ISA addresses risks associated with UF_6 and its airborne release reaction products, HF and UO_2F_2. Chapter 6.0 of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration provide 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. Appendix B of the ISA Summary and Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration identify other chemicals and typical industrial materials (e.g., acetone, solvents, acids, fuels, and oils) that are used in the ACP (including HALEU Demonstration) for assembly and maintenance activities.

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

Chemical quantities are maintained below Process Safety Management (PSM)/Risk Management Program (RMP) threshold quantities as described in Sections 6.2.2.11.1 and 6.3 of this license application.

6.2.2.11.4 Industrial Hygiene and Safety Program Managed Chemicals

Hazardous and toxic chemicals are effectively managed using IHS programs. To address these hazards, the IHS program provides the necessary protective barriers and controls that enable safe use of these chemicals in accordance with OSHA requirements (29 CFR Part 1910).

Commercial chemicals have varying toxicity and hazardous ranges and categories. Because chemicals can be used within the facilities for various purposes, the IHS program applications to chemical safety are comprehensive and are 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 ACP uses chemical review programs, program procedures, and MSDSs/SDSs. 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 Department of Energy Reservation

The Licensee subleases, from Centrus Energy Corp., certain support buildings/facilities on the DOE reservation. The ACP and the gaseous diffusion plant are separate entities for purposes of chemical safety. Each has its own chemical safety programs and shares information regarding hazardous chemicals used by the other. The DOE environmental restoration contractors and sub-contractors use the remaining reservation sectors. The DOE provides information regarding any hazardous chemicals used by these "third-parties" that could impact ACP operations. Third-party chemicals are covered by a shared site agreement and reviewed in accordance with procedures.

6.3 Requirements for New Buildings/Facilities or New Processes at Existing Facilities

System design requirements adhere to the 10 CFR 70.64 Baseline Design Criteria for chemical protection in new ACP buildings/facilities. Revision or modification to an existing 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 that become subject to the requirements of the PSM/RMP program, a pre-startup safety review is performed based on changes to the process safety information. The pre-startup safety review is an independent review to address the readiness of the system hardware, associated hazard controls, personnel (including required training), procedures, and process safety information. Records of chemical releases and documentation relating to chemical process safety are retained in accordance with Records Management and

Document Control (RMDC) requirements described in Section 11.7.1.5 of this License Application to ensure compliance with NRC's chemical process safety requirements.

6.4 References

1. 29 CFR Part 1910, *Occupational Safety and Health Standards*
2. 29 CFR 1910.119, *Process Safety Management of Highly Hazardous Chemicals*
3. 29 CFR 1910.120, *Hazardous Waste Operations and Emergency Response*
4. 40 CFR Part 68, *Chemical Accident Prevention Provisions*
5. LA-3605-0003, *Integrated Safety Analysis Summary for the American Centrifuge Plant*
6. NR-3605-0003, *Quality Assurance Program Description for the American Centrifuge Plant*
7. NRC Information Notice No. 88-100: *Memorandum of Understanding between NRC and OSHA Relating to NRC-Licensed Facilities* (53 *Federal Register* 43950, October 31, 1988), December 23, 1988
8. NUREG-1513, *Integrated Safety Analysis Guidance Document*
9. NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications,* Revision 2
10. NUREG-1601, *Chemical Process Safety at Fuel Cycle Facilities*
11. LA-3605-0003A, *Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration*

7.0 FIRE SAFETY

The American Centrifuge Plant (ACP), including the HALEU Demonstration has provisions to provide adequate protection against fire and explosions. This chapter provides descriptions of the Fire Safety Program and fire protection systems and equipment used to ensure employee and public health and safety from fires in the ACP.

The Fire Safety Program is part of the safety program that is designed to meet the requirements established in 10 *Code of Federal Regulations* (CFR) 70.62(a). The Fire Safety Program complies with requirements established in 10 CFR 70.61, 10 CFR 70.62, and 10 CFR 70.64; and the guidance provided in NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications* (Revision 2). The Fire Safety Program addresses fire safety requirements for the ACP.

The Fire Safety Program addresses requirements for ensuring the fire protection systems and fire services supporting the ACP are adequate and maintained properly. Fire services refer to emergency and fire response services, fire inspection services, and fire testing services. As discussed in Section 1.1.8 of this license application, ACO's long-term goal is to resume commercial enrichment production consistent with market demand. The ACP design is modular, with the basic building block of enrichment capacity being a cascade of centrifuges. Modular deployment would accommodate market demand on a scalable, economical gradation. The Fire Safety Program will be implemented to support the modular deployment, such that the fire protection systems/services are in place when needed.

The next phase of enrichment production includes the deployment of a cascade of centrifuges to demonstrate production of high-assay, low-enriched uranium (HALEU) fuel for advanced reactors. The primary building/facilities directly involved in HALEU Demonstration are the X-3001 Process Building, X-3012 Process Support Building, X-7725 Recycle/Assembly Building, X-7726 Centrifuge Training and Test Facility, and X-7727H Interplant Transfer Corridor. The Licensee will notify NRC well in advance of the transition into any future phases of ACP deployment.

The ACP is comprised of buildings/facilities located on the U.S. Department of Energy's (DOE) reservation in the former Gas Centrifuge Enrichment Plant (GCEP) buildings. Additional structures will be constructed to meet the specific needs of the ACP.

Many of the buildings/facilities that comprise the ACP were designed and constructed in the 1970s and 1980s to meet the codes and standards applicable at those times. These buildings/facilities have been analyzed for fire hazards, which are discussed further in Section 7.2 of this chapter. The fire protection equipment, structural features, and fire suppression systems are designed to detect, contain, and suppress fires. The major physical components of the fire protection system include fire detection, firewater 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 chapter. Fire protection design provides for adequate protection against fires and explosions in accordance with the Baseline Design Criteria contained in 10 CFR 70.64(a) and the defense-in-depth requirements of 10 CFR 70.64(b).

The Fire Safety Program with regard to building/facility, system, and equipment design, maintains the fire protection systems in existing buildings/facilities in accordance with the codes and standards that were applicable at the time of construction and installation. New buildings/facilities meet codes and standards applicable at the time of design. Modifications to existing buildings/facilities are evaluated relative to the safety benefit that could be achieved from applying current codes and standards. Justification for any deviations from the codes and standards of record are documented in writing and approved by the Authority Having Jurisdiction (AHJ). The Configuration Management Program as described in Section 11.1 of this license application, identifies the applicable codes and standards via the system requirements documents for each building/facility. The Fire Hazard Analyses (FHA) also provide this information.

National Fire Protection Association (NFPA) 801-2020, *Standard for Fire Protection for Facilities Handling Radioactive Materials*, addresses fire protection requirements for buildings/ facilities handling radioactive materials and generally references other NFPA codes and standards dealing with each specific type of equipment or program. The daughter standards are written for general commercial facilities and may not be applicable to uranium enrichment facilities. The Fire Safety Program and the ACP were reviewed to determine applicability and level of compliance with NFPA 801 and applicable daughter standards. Some ACP buildings/facilities do not meet NFPA 801 and the applicable daughter standards because they were built or established under earlier versions or different codes and standards applicable at the time of construction and installation.

The Fire Safety Program consists of five parts to provide a defense-in-depth approach to reduce the likelihood of occurrence, consequences, and damage that results from fires. First, a number of management measures are in place to ensure the availability and reliability of the fire protection items relied on for safety (IROFS), prevent fires, and minimize the consequences and damage from fires. Second, FHAs have been performed to determine vulnerability of the ACP to fires. Third, the ACP design incorporates fire prevention and fire protection requirements. Fourth, process fire safety ensures that enrichment process hazards are properly identified and addressed to ensure the health and safety of the workforce and public. Fifth, fire protection equipment and emergency response personnel are in place to minimize the consequences and damage from fires.

7.1 Fire Safety Management Measures

Fire Safety management measures are in place to ensure that IROFS are available and reliable. This is accomplished through the following, which are described in Chapter 11.0 of this license application.

- The Configuration Management Program ensures that the ACP facilities are controlled in accordance with the baseline configuration.
- The Maintenance Program ensures that IROFS equipment is maintained and tested to ensure their reliability and availability.

- The Training and Qualification program ensures that personnel performing fire protection activities relied on for safety have the applicable knowledge and skills necessary to operate and maintain the ACP in a safe manner.

- Procedures are utilized to ensure safe operations and thorough response to upset conditions involving fires.

- Audits and assessments ensure that the Fire Safety Program is adequate and effectively implemented.

- Incident reporting and investigations are performed to identify and document fire incidents to continually improve operations and programs to ensure the health and safety of the workforce and public.

- Records are maintained and controlled to ensure that IROFS for fire protection are available and reliable.

The Fire Safety/Emergency Management Manager is responsible for the Fire Safety Program, including fire services and reports to the Production Support Manager. This manager has the authority to ensure that fire safety receives appropriate priority.

An experienced fire professional is assigned as the 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.

The specific NFPA standards applicable to the ACP are identified in Table 7.1-1 of this chapter. Any changes where full compliance with the applicable NFPA standards is not maintained will be documented and justified by the AHJ. Modifications to fire protection systems and programs are made in accordance with 10 CFR 70.72.

The Plant Safety Review Committee, as described in Chapter 2.0 of this license application, provides a review role of fire safety at the ACP. The membership, structure, and responsibilities of this multi-discipline committee are defined in a plant procedure. The procedure includes the responsibility to review fire safety issues and to integrate changes to the plant with adequate consideration of fire safety.

The ACP Fire Safety Program management measures are grouped into four areas:

- Fire prevention;

- Inspection, testing, and maintenance of fire protection systems;

- Emergency response organization qualifications, drills, and training; and

- Pre-fire plans.

7.1.1 Fire Prevention

Fire prevention is a program across the ACP to minimize the potential for an incipient fire. The following are the major points that are addressed by the program.

- Workers are required to review and understand fire safety information including fire prevention procedures, emergency alarm response, and fire reporting. Documented building/facility inspections are conducted periodically and remedial actions are taken when conditions of concern are identified (i.e., accumulation of unnecessary transient combustibles, the presence of uncontrolled ignition sources, or obstruction of egress routes).

- General housekeeping practices and control of transient combustibles are established.

- Control of flammable and combustible liquids and gases is handled in accordance with the NFPA 30–2018, *Flammable and Combustible Liquids Code* and NFPA 55-2020, *Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tank.*

- Ignitions sources are controlled.

- Fire reports documenting fire investigation and corrective actions are documented through the Corrective Action Program as described in Section 11.6 of this license application.

- Smoking is restricted to designated areas outside of buildings/facilities.

- Construction activities are performed in a manner that meets the requirements of NFPA 241-2019, *Standard for Safeguarding Construction, Alteration, and Demolition Operations*.

7.1.1.1 Control of Impairment to Fire Protection Systems

Impairment of fire detection, fire alarms, and fire barriers requires notification to the building custodian of the reason for the impairment, the specific impairment, the expected duration of the impairment, and system restoration time. Compensatory actions are initiated when detection, alarms, or barriers are out of service and 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.

Closure of ACP 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 the Fire Safety/Emergency Management Manager have the authority to issue permits and operate fire protection valves.

The ACP firewater permit system provides for notification to the building custodian of the reason for the impairment, the expected duration of the impairment, system restoration time, and residual partial system impairment (e.g., branch line removed). Compensatory actions are initiated when building sprinkler systems are out of service and 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. ACP systems taken out of service for repair are usually returned to service within an eight-hour period; however, the extent of the actual repairs will affect completion time.

7.1.1.2 Hot Work Permits

Hot work is controlled by procedure complying with NFPA 51B-2019, *Standard for Fire Protection During Welding, Cutting, and Other Hot Work* and applicable Occupational Safety and Health Administration (OSHA) requirements per 29 CFR Part 1910.252 The permit system ensures that cutting, welding, and other hot work conducted in plant areas not normally used for such purposes will be conducted utilizing a permit system/process and performed in a manner that is consistent with industry fire prevention practices. This includes pre-job inspection, stationing a fire watch during the hot work as required, and post-job fire watch to prevent delayed ignition of any combustibles.

Selected managers and supervisors are trained and authorized to write hot work permits. Personnel performing fire watches receive additional training. 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.

7.1.2 Inspection, Testing, and Maintenance

Fire protection equipment is inspected and tested upon installation in accordance with NFPA 25-2002 *Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*. Periodic inspection and testing of fire protection equipment are performed by or overseen by trained personnel to help ensure that fire safety related IROFS are available and reliable. The testing and inspection of equipment is performed in accordance with procedures that include test frequencies as defined by the Fire Safety/Emergency Management Manager. The major elements of the plant inspection program are identified as follows.

- Flow test sprinkler systems
- Test manual fire alarms (pull stations)
- Test sprinkler water flow alarms
- Test supervisory alarm devices including control valves, low air pressure, low temperature, and loss of power
- 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
- Inspect sprinkler systems risers
- Inspect portable fire extinguishers

7.1.3 Emergency Response Organization Qualifications, Drills, and Training

The ACP relies upon a qualified provider to perform emergency response to fire and other types of accident scenarios occurring at the ACP. Employees receive initial and biennial fire safety training as part of General Employee Training (GET) on emergency preparedness. This includes emergency reporting, building/facility evacuation, and fire extinguisher familiarization. GET is described in Section 11.3.1.1 of this license application.

A qualified supplier provides fire department response to an emergency. This supplier is staffed, trained, and equipped adequately to meet the needs of the ACP and the commitments contained in this license application. The qualified provider will have adequate resources to meet the needs of the ACP. This requires appropriately trained and qualified firefighting personnel, available 24-hours per day, as well as a minimum complement of equipment. There will be a minimum of four qualified fire fighters and one supervisor available to respond per shift. These four fire fighters cover entry and backup (two each). Equipment requirements include one pumper truck with a minimum capacity of 1,000 gpm, one ambulance, and one HAZMAT truck with radiological and rescue equipment. The time to apply water onto a fire will not exceed 20 minutes, 90 percent of the time. This is assured through assessments performed in accordance with Section 11.5 of this license application that confirms that the level of service is consistent with performance requirements specified in a letter of agreement.

Firefighter training is equivalent to the state certified firefighter training curriculum. Emergency medical response personnel meet requirements for state certification as emergency medical technicians and are 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. Training records are kept of the training activities. Training is based on national standard emergency response methodology with plant-specific training on issues unique to the plant. Specific training activities include firefighting, hazardous material response, confined space rescue, emergency medical response, radiological emergencies, and rescue. Drills are conducted as part of the plant emergency plan.

7.1.4 PreFire Planning

Prefire plans are developed as part of the building emergency packet. Pre-fire plans for HALEU Demonstration include the following buildings and areas: X-3001 Process Building; X-3012 Process Support Building; X-7725 Recycle/Assembly Building; X-7726 Centrifuge Training and Test Facility; and X-7727H Interplant Transfer Corridor. Pre-fire plans for other facilities will be developed prior to deployment of operations involving licensed materials in those facilities.

Each pre-fire plan contains the following applicable information about the building or area:

- Facility description/construction,
- Specific hazards to emergency responders,
- Search and rescue considerations,
- Fire protection equipment/systems available,
- Utility shut-offs/start-ups,
- Fire loading concerns,
- Unique fire fighting strategy and tactics,
- Fire extension concerns, and
- Ventilation methodology.

Trained personnel review these pre-fire plans as part of the building inspection. As buildings are modified to meet the changing operations, the pre-fire plans are scheduled for review and updates to assure the revised conditions are addressed. As new buildings are added to meet the changing operations, pre-fire plans will be developed prior to placing the buildings in operation.

Table 7.1-1 Applicable National Fire Protection Association Codes and Standards

Code No.	Title	Revision
NFPA 10	*Standard for Portable Fire Extinguishers*	2018
NFPA 13	*Standard for the Installation of Sprinkler Systems*	2019
NFPA 15	*Standard for Water Spray Fixed Systems for Fire Protection*	2017
NFPA 25	*Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*	2002
NFPA 30	*Flammable and Combustible Liquids Code*	2018
NFPA 51B	*Standard for Fire Prevention During Welding, Cutting, and Other Hot Work*	2019
NFPA 55	*Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks*	2020
NFPA 70	*National Electric Code*	2005
NFPA 72	*National Fire Alarm Code*	2002
NFPA 75	*Standard for the Protection of Electronic Computer/Data Processing Equipment*	2003
NFPA 80	*Standard for Fire Doors and Fire Windows*	1999
NFPA 101	*Life Safety Code*	2018
NFPA 220	*Standard on Types of Building Construction*	1999
NFPA 232	*Standard for the Protection of Records*	2000
NFPA 241	*Standard for Safeguarding Construction, Alternation, and Demolition Operations*	2019
NFPA 801	*Standard for Fire Protection for Facilities Handling Radioactive Materials*	2020

7.2 Fire Hazards Analysis

FHAs have been performed for the following buildings and areas; X-3001, X-3002, X-3012, X-7725, X-7726, X-7727H, X-3344, X-3346, X-3346A, X-745G-2, X-7746S, and X-7746W. The FHAs applicable to HALEU Demonstration include those for X-3001, X-3012, X-7725, X-7726, and X-7727H. The FHAs and supporting analyses ensure that the fire prevention and fire protection requirements have been evaluated and incorporated. The analyses consider the building's/facility's specific design, layout, and anticipated operating needs and considers acceptable means for separation or control of hazards, the control or elimination of ignition sources, and the suppression of fires. FHAs for other facilities will be developed and implemented prior to deployment of operations involving licensed materials in those facilities.

The FHAs and supporting analyses were used in the Integrated Safety Analysis (ISA) for the ACP to determine the credible fire accident scenarios, their likelihood of occurrence, the associated consequences, and the necessary IROFS to reduce the likelihood of occurrence and/or the consequences to meet performance requirements. The results of the ISA are presented in the ISA Summary for the American Centrifuge Plant and Addendum 1 of the ISA Summary for the American Centrifuge Plant - HALEU Demonstration.

To ensure an adequate level of safety is maintained, fire hazards for each of the buildings are evaluated periodically and documented in a building survey. The building survey results are used to update the FHAs and ISA as necessary. Further discussion of the FHA, ISA, and building survey approaches are described below.

For new buildings or facilities, FHAs are performed during the design development process to ensure that the fire prevention and fire protection requirements have been evaluated and incorporated into the design. The analysis considers the facility's specific design, layout, and anticipated operating needs and considers acceptable means for separation or control of hazards, the control or elimination of ignition sources, and the suppression of fires.

7.2.1 Fire Hazards Analysis Approach

Fire Hazards Analyses provide a general description of the physical characteristics of the buildings/facilities that outlines the fire prevention and fire protection systems to be provided. A FHA defines the fire hazards that can exist, and states the loss limiting criteria to be used in the design of a building and/or facility. FHAs provide a formal review and periodic evaluation of the occupancy and the fire protection associated with a building/facility and includes the following elements:

- A listing of the codes and standards is used for the design of the fire protection systems, including the published standards of NFPA.

- The FHA defines and describes the characteristics associated with potential fires for areas that contain combustible materials, such as fire loading, hazards of flame spread, smoke generation, toxic contaminants, and contributing fuels.

- The FHA lists the fire protection system criteria and the criteria to be used in the basic design for such items as water supply, water distribution systems, and fire pump supply.

- The FHA describes the performance criteria for the detection systems, alarm systems, automatic suppression systems, manual systems, chemical systems, and gas systems for fire detection, confinement, control, and extinguishment.

- The FHA describes the design for suppression systems and for smoke, heat, and flame control; combustible and explosive gas control; and toxic and contaminant control as necessary. The FHA also describes the operating functions of the ventilating and exhaust systems to be used during the period of fire extinguishment and control.

- The FHA uses the features of building and facility arrangements and the structural design features to generally define the methods for fire prevention, fire extinguishing, fire control, and control of hazards created by fire. Fire barriers, egress, firewalls, and the isolation and containment features provided for flame, heat, hot gases, smoke, etc., are also addressed.

- The FHA identifies the dangerous and hazardous combustibles and the maximum quantities estimated to be present in the building/facility. The FHA also identifies where these materials can be located appropriately in the building/facility.

- Based on the expected quantities of combustible materials, the types of potential fires, their estimated severity, intensity, duration, and the potential hazards created for each fire scenario reviewed, the probable and possible maximum losses from fires are described in the FHAs.

- Where safe shut down of safety related equipment is necessary, the FHA will define the essential electric circuit integrity needed during fire, and evaluates the electrical and cable fire protection; the fire confinement control; and the fire extinguishing systems that will be needed to maintain their integrity.

- The FHA evaluates life safety, protection of critical process/safety equipment, lightning protection, provision to limit contamination, potential for radioactive release, and restoration of the building/facility after a fire.

7.2.2 Integrated Safety Analysis

An ISA of the design, construction, and operation of the commercial ACP 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 associated Addendum to the ISA Summary was also performed for the HALEU Demonstration applying the same guidance and process. The ISA contains the following 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.

A number of the release scenarios evaluated in the ISA have an explosion or fire as the initiating event and are evaluated for the FHAs. The ISA determines the likelihood of occurrence for the fire scenarios and resulting consequences associated with the release of uranium hexafluoride (UF_6) and its airborne release reaction product, hydrogen fluoride (HF) assuming the fire is unmitigated. Then the ISA identifies IROFS and related management measures necessary to prevent the accident and/or mitigate the consequences in accordance with the performance criteria in 10 CFR 70.61. This information is presented in the ISA Summary for the American Centrifuge Plant and Addendum 1 of the ISA Summary for the American Centrifuge Plant - HALEU Demonstration.

UF_6 is the primary hazardous material in the commercial ACP operation and HALEU Demonstration 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.

7.2.3 Building Surveys

The building surveys are conducted, in accordance with written procedures on a periodic basis, to ensure the buildings/facilities, systems, and operations continue to meet the codes and standards to which they were built and operated, and do not violate any safety basis that were established in the ISA for the credible accident scenarios. The building surveys also ensure no new credible fire scenarios have been created.

7.3 Building/Facility Design

There are fire hazards related to the enrichment process. Fire hazards are typical industrial hazards, including maintenance; incidental use of chemicals and flammable liquids; and energized electrical equipment in the buildings. Accident potentials are discussed in the FHAs and ISA.

The ACP buildings/facilities are large and spread across the DOE reservation, which minimizes the effects that a fire or explosion could have on adjacent buildings and operations. Ventilation supply and exhaust locations are considered with regard to contamination potential and smoke control.

The primary ACP buildings/facilities are X-3001, X-3002, X-3012, X-3344, X-3346, X-3346A, X-7725, X-7726 buildings/facilities, and X-7727H corridor. Only X-3001, X-3012, X-7725, X-7726, and X-7727H are used for HALEU Demonstration. The X-3001, X-3002, X-3012, X-3344, X-3346, X-3346A, X-7725, X-7726 buildings/facilities, and X-7727H corridor 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. Each building is considered a single fire area with exception of the X-3346, X-7725, X-7726 buildings/facilities, and X-7727H corridor. Sprinkler coverage is provided in each building/facility. The sprinkler and water systems are described below. There are no water-exclusion areas in the ACP. Combustible loading is typically low and the fire hazards are limited to normal industrial activities. Exceptions are identified in the building survey report or by the building/facility manager. These include such things as electrical switchgear and transformers, and maintenance activities.

Use of firewater and potential firewater accumulation has been reviewed in each of the buildings/facilities to assure no unsafe accumulations can occur with regard to criticality, equipment loss, or spontaneous combustion.

Firewater runoff to the environment is controlled by the presence of holding ponds that can reduce or terminate releases as necessary to minimize environmental impact. There are no credible accident scenarios that could result in a criticality event in the holding ponds.

As indicated previously, the X-3001, X-3002, X-3012, X-3344, X-3346A, X-745G-2, X-7746S, and X-7746W are each considered single fire areas, but the X-7725 building and X-7726 facility, and X-7727H corridor are considered as a single fire area and the X-3346 building is considered as two fire areas (Feed Area and Withdrawal Area). Fire areas are considered to be any location bounded by fire rated construction with a minimum rating of two hours and equivalently fire rated doors, dampers, or penetration seals. Building and area separation is used as a method of limiting fire spread. The X-7725 and X-3001 buildings are, connected by the X-7727H corridor, of the same construction. Each are protected by automatic sprinkler system and have acceptable amounts of combustibles.

Review of the emergency egress paths for the existing buildings/facilities is accomplished using NFPA 101-2018, *Life Safety Code,* as guidance. Some buildings do not comply with the travel distances due to their size. Exit arrangements are adequate because of the low occupancy levels, low combustible loading, large number of exits, and fixed fire suppression systems in the buildings.

Combustible storage in the buildings is considered as part of the hazard evaluation described in Section 7.2 of this chapter. There are no significant quantities of flammable liquids or gases used in the enrichment process; however, centrifuge component manufacturing may be performed in the X-7725 and involve significant quantities of flammable liquids. The use of these liquids and incidental use of other flammable liquids and gases is controlled in accordance with NFPA 30-2018, *Flammable and Combustible Liquids Code* and NFPA 55-2020, *Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks.*

Electrical systems are installed in accordance with NFPA 70-2005, *National Electric Code.*

ACP building/facility design elements include fire protection lighting and fire barriers to ensure personnel safety in accordance with the applicable NFPA identified in Table 7.1-1.

Security provisions to maintain control of classified material during fire events are addressed in the *Security Plan for the Protection of Classified Matter for the American Centrifuge Plant* (Reference 22).

New buildings/facilities are designed, constructed, and operated to meet the codes and standards applicable at the time of design development.

The Cylinder Storage Yards X-7746S, and X-7746W) have fire hydrants equipped with monitor nozzles. Workers are trained to initiate the nozzles should a fire occur within the yards.

7.3.1 Fire Suppression Systems

Fire Suppression is provided in the following buildings/facilities to support HALEU Demonstration: X 3001, X-3012, X-7725, X-7726, and X-7727H. Fire suppression is provided in the following buildings/facilities to support commercial ACP operation: X-3001, X-3002, X-3012, X-3344, X-3346, X-3346A, X-7725, X-7726 buildings/facilities, and X-7727H corridor is provided by sprinkler systems. The systems are hydraulically designed and installed to meet or exceed the NFPA recommended sprinkler densities for Ordinary Hazard Group 1 occupancies and storage 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. This design is sufficient to ensure that credible fire related accident scenarios can be controlled given the building designs, equipment layout, and anticipated combustible loadings. Fire Suppression in other buildings/facilities (e.g., X-3002, X-3346, X 3346A) will be provided prior to deployment of operations involving licensed materials in those facilities.

Existing suppression systems are maintained in accordance with the applicable codes and standards enforced at the time of construction and installation. New suppression systems will meet NFPA 13-2019, *Standard for the Installation of Sprinkler Systems* and NFPA 25-2017, *Standard for Water Spray Fixed Systems for Fire Protection.* When modifying existing buildings/facilities, the safety benefit from applying current codes and standards will be evaluated to determine if the change is justified. The evaluation and decision made will be documented.

7.3.2 Fire Alarms

The sprinkler systems are connected to the Fire Alarm system. This system meets the requirements 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, X-3002, X-3012, X-3344, X-3346, X-3346A, X-7725, X-7726 buildings/facilities, and X-7727H corridor. Alarms are received in the X-300 Plant Control Facility and X-1007 Fire Station. Alarm announcement is not local, but building evacuation can be manually initiated from the X-3012 Area Control Room, X-300, and X-1007 or locally in some areas.

7.4 Process Fire Safety

The ACP has addressed process fire safety through the design of the buildings and operations such that consideration is taken for fire hazards that may be present in order to protect the workforce and public. Hazardous areas are identified to ensure the workforce is cognizant of hazardous material and operations. The ISA has been performed to identify the credible accident scenarios and establish the necessary IROFS to ensure the health and safety of the workforce and public. The IROFS for the ACP are identified in the ISA Summary for the commercial ACP operation. The IROFS for HALEU Demonstration are identified in Addendum 1 of the ISA Summary. IROFS associated only with the commercial ACP operation will be implemented prior to deployment of operations involving licensed materials in those facilities.

The ACP buildings/facilities are designed in accordance with the codes and standards as identified in Section 7.1 above. The ACP hazardous areas are identified as part of the pre-fire plans required in Section 7.1.4 above. The ACP ISA is discussed in Section 7.2.2 of this chapter and Chapter 3.0 of this license application.

The ISA determines the likelihood of occurrence for the explosion and fire scenarios and resulting consequences associated with the release of UF_6 and its airborne release reaction product, HF assuming the accident is unmitigated. The ISA identifies IROFS and related management measures necessary to prevent the accident and/or mitigate the consequences in accordance with the performance criteria in 10 CFR 70.61. The IROFS identified by the ISA to prevent or mitigate explosion and fire related scenarios are grouped in the following three categories.

- Combustible Material Control
- Fire Suppression and Response
- Fire/Explosion Prevention

UF_6 is the primary hazardous material in the ACP. In the presence of moist air, UF_6 reacts to form HF gas and UO_2F_2. The ISA considers UO_2F_2 for radiological and toxicological hazards and HF for toxicological hazards. Other chemicals evaluated are activated alumina pellets used in the alumina traps to filter UF_6 gas, compressed gases (e.g., nitrogen, acetylene), perfluorocarbon fluid used in the equipment brine heating/cooling system, other refrigerants used in the various process refrigeration systems, janitorial supplies, fire extinguishing agents, and non-flammable oils used within the centrifuge upper and lower support assemblies. These other chemicals are not considered to have a significant hazardous interaction capability.

If centrifuge component manufacturing is performed within the ACP, additional materials are required for the process that will present fire safety and health concerns. These additional materials include carbon fibers, resin systems (resins, hardeners, and modifiers), prepregs (fibers/resin system) and for cleaning chemicals such as acetone, alcohols, carbon dioxide, ethanol, and Freon 134.

7.5 Fire Protection and Emergency Response

The design and operation of the buildings/facilities are evaluated on a periodic basis to ensure fire hazards are controlled. Fire protection systems are present to further reduce the risk of fires that could result in a release of hazardous material. Emergency response is provided to add defense-in-depth to the fire protection systems and respond to areas where fire protection systems do not exist.

7.5.1 Fire Protection Engineering

Fire protection engineering support is available to evaluate fire hazards; review changes to maintenance and process systems; and provide in-house consultation under the direction of the Fire Safety/Emergency Management Manager. They also perform the building surveys as described in Section 7.2.3 of this chapter.

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, building/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.

Reported fires are investigated using a graded approach through the Corrective Action Program. This includes investigations by fire officers, engineers, or by multidiscipline teams as warranted. Results of investigations are considered for distribution throughout ACP operations to prevent future reoccurrences. Details of incident investigation in the ACP are described in Section 11.6 of this license application.

7.5.2 Alarm and Fixed Fire Suppression Systems

The ISA credits fire suppression to ensure that credible fire accident scenarios do not result in consequences that would exceed the performance criteria established in 10 CFR 70.61. The alarm and fire suppression systems are designed and installed with adequate capabilities to detect and suppress the credible accident scenarios identified by the ISA. The alarm and fixed fire suppression relied on for HALEU Demonstration is identified in Addendum 1 of the Integrated Safety Analysis – HALEU Demonstration.

The firewater supply to support fire suppression systems is provided by the DOE reservation system. The firewater supply is sufficient to meet the anticipated needs of the ACP. To ensure the firewater is available and reliable, assessment requirements of Section 11.5 of this license application are performed. See Section 7.5.3 of this chapter.

Fire detection is based upon heat and is an integral part of the fire suppression systems. Fire suppression systems have sprinkler heads with fusible links or gas expansion actuators to initiate water flow when specific temperatures are reached. Water flow alarms on the fire suppression systems provide fire detection. System flow is monitored to provide alarms for emergency response.

The fire alarm system monitors fire suppression systems in the ACP buildings. Alarms caused by non-fire conditions (i.e., spurious water flow alarms from pressure surges) are reviewed by fire safety personnel and identified for maintenance as needed. The system includes alarm notification to the X-300 and X-1007. Alarm rooms are manned as necessary to support prompt notification of emergency response personnel to investigate and respond to alarm conditions.

Manual pull stations are located throughout the buildings/facilities to provide additional alarm capability. Operation of a pull station initiates an alarm at the central alarm receiving locations (X-1007 and X-300buildings) but is not announced locally.

The ACP has evacuation alarm initiation capability in areas that can be initiated locally, in addition to remote initiation capability from the X-3012 Area Control Room, X-300, and X-1007 buildings.

Fixed automatic fire suppression systems provide the means of detection, control, and suppression of fires at the ACP. These fixed fire suppression systems are inspected, tested, and maintained on a regular basis in accordance with approved procedures.

7.5.3 Firewater Distribution System

The ACP fire suppression systems are part of the DOE reservation firewater distribution system. This system is capable of supplying firewater at rates and durations adequate to meet the anticipated needs of the ACP. The firewater distribution system is an underground piping system laid out such that each ACP building/facility can be supplied from at least two sources. The fire hydrants adjacent to ACP buildings/facilities are also supplied by the firewater distribution system. Additional components that support firewater distribution of the firewater storage tanks and firewater pumps.

The firewater storage tanks include one 300,000-gallon elevated tank and two 2,000,000-gallon surface tanks. The firewater pumps include two electric pumps and one diesel pump each with a capacity to pump up to 4,000 gallons per minute. The diesel pump has enough fuel to run for the durations needed to meet the anticipated needs of the ACP.

7.5.4 Mobile and Portable Equipment

Mobile and portable fire protection equipment are provided by a qualified supplier. Portable fire extinguishers are available throughout the ACP. Size, selection, and distribution of extinguishers are determined in accordance with NFPA 10-2018, *Standard for Portable Fire Extinguishers*.

7.5.5 Emergency Response

The ISA credits emergency response to ensure that credible fire accident scenarios do not result in consequences that would exceed the performance criteria established in 10 CFR 70.61.

Fire department emergency response is provided by a qualified supplier. This supplier is staffed, trained, and equipped adequately to meet the needs of the ACP. See Section 7.1.3 of this chapter. ACP workers are trained as indicated in Section 11.3 of this license application to recognize emergency conditions and alert the emergency response group.

7.5.6 Control of Combustible Materials

The ISA credits combustible materials control programs inside and outside the ACP buildings/facilities to ensure that credible fire accident scenarios do not result in consequences that would exceed the performance criteria established in 10 CFR 70.61. This covers the ACP primary facilities and is addressed on a continuous basis by the building/facility custodians. It also includes limited use of fossil fuel and other combustible material. Combustible materials control is assured through training and procedures as discussed in Sections 11.3 and 11.4 of this license application.

7.5.7 Use of Noncombustible Materials

The ISA credits use of noncombustible materials in the construction and operation of the ACP buildings/facilities to ensure that credible fire accident scenarios do not result in consequences that would exceed the performance criteria established in 10 CFR 70.61. This includes use of construction material such as concrete, steel, insulation, and refrigerant. Use of noncombustible materials is assured through the Configuration Management Program discussed in Section 11.1 of this license application.

7.5.8 Control of Combustible Mixtures

The ISA credits control of combustible gases and mixtures in the construction and operation of the ACP buildings/facilities and manufacture of equipment to ensure that credible fire accident scenarios do not result in consequences that would exceed the performance criteria established in 10 CFR 70.61. Control of combustible mixtures is assured through the Maintenance Program discussed in Section 11.2 of this license application.

7.5.9 Placement of Equipment and Operations

The ISA credits placement of equipment in ACP buildings/facilities to ensure that credible fire accident scenarios do not result in consequences that would exceed the performance criteria established in 10 CFR 70.61. Proper placement of equipment and operations is assured through the Configuration Management Program discussed in Section 11.1 of this license application.

7.6 References

1. 29 CFR Part 1910.252, *Occupational Safety and Health Standards*

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

3. NFPA 10-2018, *Standard for Portable Fire Extinguishers*

4. NFPA 13-2019, *Standard for the Installation of Sprinkler Systems*

5. NFPA 15-2017, *Standard for Water Spray Fixed Systems for Fire Protection*

6. NFPA 25-2002, *Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*

7. NFPA 30-2018, *Flammable and Combustible Liquids Code*

8. NFPA 51B-2019, *Standard for Fire Prevention During Welding, Cutting, and Other Hot Work*

9. NFPA 55-2020, *Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks*

10. NFPA 70-2005, *National Electric Code*

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

12. NFPA 75-2003, *Standard for the Protection of Electronic Computer/Data Processing Equipment*

13. NFPA 80-1999, *Standard for Fire Doors and Fire Windows*

14. NFPA 101-2018, *Life Safety Code*

15. NFPA 220-1999, *Standard on Types of Building Construction*

16. NFPA 232-2000, *Standard for the Protection of Records*

17. NFPA 241-2019, *Standard for Safeguarding Construction, Alteration, and Demolition Operations*

18. NFPA 801-2020, *Standard for Fire Protection for Facilities Handling Radioactive Materials*

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

20. NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications*, Revision 2

21. LA-3605-0003A, *Addendum 1 of the ISA Summary for the American Centrifuge Plant – HALEU Demonstration*

22. SP-3605-0041, *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant*

8.0 EMERGENCY MANAGEMENT

As discussed in Section 1.1.8 of this license application, it is the long-term goal of the Licensee to deploy the American Centrifuge Plant (ACP) in a modular fashion on a scalable, economical gradation consistent with market demand. American Centrifuge Operating, LLC (ACO), the Licensee, would develop and submit future license amendments to allow additional phases of modular deployment up to the currently U.S. Nuclear Regulatory Commission (NRC)-approved full capacity operation of 3.8 million separative work units. Pursuant to 10 *Code of Federal Regulations* (CFR) 70.22(i), the Licensee developed an NRC-approved Emergency Plan for the fully deployed ACP and other on-going activities on the U.S. Department of Energy (DOE) reservation in Pike County Ohio. The previously NRC-approved plan conforms to the Regulatory Guide 3.67, *Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities*, dated January 1992. Although not required or implemented for the High Assay Low Enriched Uranium (HALEU) Demonstration Program, the Emergency Plan will support future ACP deployment phases.

The information documented in the previously NRC-approved emergency plan includes: 1) description of the facility; 2) summary credible emergencies; 3) classification and notification of accidents; 4) responsibilities; 5) emergency response measures; 6) equipment and facilities designated for use during emergencies; 7) methods for maintaining emergency preparedness; 8) emergency records and reports; 9) recovery and restoration measures; and 10) a commitment to comply with the *Community Right-To-Know Act*.

The previously NRC-approved plan remains as part of this license application as document NR-3605-0008, *Emergency Plan for the American Centrifuge Plant* in Piketon, Ohio. The Licensee would notify the NRC well in advance of the transition into any future phases of deployment that would require use of this previously NRC-approved NR-3605-0008.

8.1 High Assay Low Enriched Uranium Demonstration

No Emergency Plan as discussed under 10 CFR 70.22(i) is needed for the HALEU Demonstration Program. DAC-3901-0005, *Evaluation of No Need for an Emergency Plan for the HALEU Demonstration*, provides the evaluation stipulated in 10 CFR 70.22(i)(1)(i) to demonstrate that no Emergency Plan is required for the HALEU Demonstration Program. The evaluation shows that the maximum dose to a member of the public offsite due to a release of radioactive materials would not exceed 1 roentgen equivalent man (rem) effective dose equivalent or an intake of 2 milligrams (mg) of soluble uranium (U).

Fluor BWXT Portsmouth, LLC (FBP); Portsmouth Mission Alliance, LLC; and Mid-America Conversion Services, LLC are the DOE's primary contractors at the DOE Portsmouth site. FBP currently serves as the Decontamination and Decommissioning contractor and provides emergency response capabilities at the site compliant under DOE Order 151.1D, *Comprehensive Emergency Management System*. Through a reverse work authorization arrangement, FBP provides emergency response to the ACP. With augmentation and coordiation with ACO personnel where appropriate, FBP provides the following:

- Emergency Response Organization
- Emergency Facilities and Equipment/Systems
- Fire Department Response*
- Emergency Operations Center (EOC)
- Alternate EOC
- Joint Information Center
- Plant Shift Superintendent/Incident Command Support
- Emergency Medical Support
- Offsite Response Interfaces
- Announcement of Protective Actions
- Emergency Public Information
- Communications and Notifications, as appropriate
- Consequence Assessment
- Support with Termination and Recovery, as appropriate
- Support with Coordinating and Assessing Readiness Assurance

* FBP operates and maintains the X-1007 Fire Station on the DOE reservation. This is a 24 hours a day/7 days a week dedicated fire department which has minimum staffing requirements to maintain appropriate manpower for emergency response on the DOE reservation. Fire personnel are certified as State of Ohio Level II (Professional Firefighters) and minimum State of Ohio Emergency Medical Technicians with one Paramedic per shift.

8.1.1 Nuclear Criticality

The primary radiation alarm system is the Criticality Accident Alarm System (CAAS), designed to detect a nuclear criticality and provide annunciation by audible evacuation alarms that are supplemented by visual alarms in some areas, such as high-noise areas that will alert personnel to evacuate the immediate area.

Operations involving fissile material are evaluated for Nuclear Criticality Safety (NCS) considerations prior to initiation. The need for CAAS coverage is considered during the evaluation process. CAAS coverage is provided, unless it is determined that coverage is not required per the requirements of 10 CFR 70.24 and the finding is documented in an NCS Evaluation. CAAS coverage is provided for HALEU Demonstration fissile material operations.

The CAAS is designed to detect gamma radiation levels that would result from the minimum criticality accident of concern as defined in 10CFR70.24(a)(1). The CAAS is designed to provide annunciation by audible alarms that are supplemented by visual alarms in some areas, such as in high-noise areas.

The criticality detection system consists of detector clusters and an alarm system. When a criticality accident alarm activates, a radiation alarm is generated actuating building local horns. Alarm activation requires evacuation of personnel from the affected area to a designated monitoring station that is located a minimum evacuation distance of 125 ft from the facility with the active CAAS alarm. Trained emergency responders are dispatched to the facility evacuation point to provide evacuees and Incident Command with additional guidance, as appropriate. Based on the alarm location, Incident Command can direct the actions necessary to respond to the accident in coordination with technical personnel. The EOC is activated and provides coordinated support for the response. Emergency response to CAAS alarms and/or nuclear criticality events is consistent with guidance contained in ANSI/ANS-8.23-2007, *Nuclear Criticality Accident Emergency Planning and Response*.

Coordinated response exercises and local drills are performed periodically to familiarize personnel with proper response actions and assembly locations.

8.2 References

1. American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.3-1997, *Criticality Accident Alarm System*

2. American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.23-2007, *Nuclear Criticality Accident Emergency Planning and Response*

3. Regulatory Guide 3.67, *Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities*, Revision 1

4. NR-3605-0008, *Emergency Plan for the American Centrifuge Pla*nt

5. DAC-3901-0005, *Evaluation of No Need for an Emergency Plan for the HALEU Demonstration*

6. NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications,* Revision 2

7. DOE Order 151.1D, *Comprehensive Emergency Management System*

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9.0 ENVIRONMENTAL PROTECTION

The American Centrifuge Plant (ACP) is located in Piketon, Ohio on the U.S. Department of Energy (DOE) reservation, adjacent to the former U.S. Nuclear Regulatory Commission (NRC) regulated Portsmouth Gaseous Diffusion Plant (GDP). The ACP Environmental Protection Program is modeled after the well-seasoned GDP environmental protection program. The ACP program thus takes advantage of the well-established programmatic elements and experience and many years of existing environmental data. This approach will provide maximum protection to the public and the environment. The Production Support Manager is responsible for the ACP Environmental Protection Program. Details of the minimum requirements for the managers and staff supporting the Environmental Protection Program are provided in Chapters 2.0 and 11.0 of this license application.

As discussed in Section 1.1.8 of this license application, American Centrifuge Operating, LLC's (ACO) long-term goal is to resume commercial enrichment production consistent with market demand. The ACP design is modular, with the basic building block of enrichment capacity being a cascade of centrifuges. Modular deployment would accommodate market demand on a scalable, economical gradation. As such, the Environmental Protection Program will be implemented to support the modular deployment.

The next phase of enrichment production includes the deployment of a cascade of centrifuges to demonstrate production of high-assay, low-enriched uranium (HALEU) fuel for advanced reactors. The primary building/facilities directly involved in HALEU Demonstration are the X-3001 Process Building, X-3012 Process Support Building, X-7725 Recycle/Assembly Building, X-7726 Centrifuge Training and Test Facility, and X-7727H Interplant Transfer Corridor. The Licensee will notify NRC well in advance of the transition into any future phases of ACP deployment. For further plant and process specifics related to the HALEU Demonstration Program, refer to LA-3605-0003A, *Addendum 1 of the ISA for the American Centrifuge Plant – HALEU Demonstration.*

The general use of the term ACP in the remainder of this chapter is intended to refer to both the commercial ACP operation and the HALEU Demonstration. HALEU Demonstration will be specifically noted, as necessary, when the context is uniquely applicable to HALEU Demonstration.

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 update to the Environmental Report for the American Centrifuge Plant meeting the requirements of 10 CFR 51.45 was prepared and is submitted for review as part of this license application as document LA-3605-0002, *Environmental Report for the American Centrifuge Plant*, Revision 17.

9.2 Environmental Protection Measures

9.2.1 Radiation Protection Program

The ACP Environmental Radiation Protection Program is based on 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 legal limits.

- It is the responsibility of each employee to conduct their 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 (As Low As Reasonably Achievable) Goals for Effluent Control

The ACP 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 DOE reservation boundary and to which any member of the public has unrestricted access.

The ALARA goal for airborne radioactive releases from the ACP is five percent of the NRC constraint (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, calculated as described in Section 9.2.2.1.2. This is also less than 15 percent of the most restrictive limit under 40 CFR Part 190, based on site experience.

The ALARA goal for waterborne radioactive releases from the ACP is ten percent of the airborne ALARA goal, or an annual TEDE of 0.05 mrem to the most exposed member of the public. This is equivalent to 0.05 percent of the 10 CFR 20.1301 limit on annual public dose. This goal is based on the assumption that: 1) the effluent limits in 10 CFR Part 20, Appendix B, Table 2 are equivalent to an annual public dose of 50 mrem; and 2) maximum public exposure occurs in the Scioto River with a dilution factor of at least 100:1. The principal liquid effluent stream from the ACP discharges directly to the river via a buried pipeline and the actual dilution factor between site effluents and the Scioto River is on the order of 5,000:1. Consequently, the second assumption should be very conservative.

The ACP also establishes Baseline Effluent Quantities (BEQs) for each monitored vent and monitored outfall and compares measured weekly effluents to these BEQs. Weekly effluents that are less than the BEQs cannot approach the dose limit in 10 CFR 20.1301 or the dose constraint in 10 CFR 20.1101. Weekly effluents that are not less than the applicable BEQs are evaluated as described in Sections 9.2.2.1.3 and 9.2.2.2.3 of this chapter, to determine whether they may cause the ACP to exceed regulatory limits or the ALARA goals. Notifications and corrective actions are implemented as described in those sections and Table 9.2-1.

9.2.1.2 Effluent Controls

9.2.1.2.1 Control of Airborne Effluents

<u>X-3346 Feed and Withdrawal Building</u>

The X-3346 operations are applicable to commercial ACP operations only and are not used in the HALEU Demonstration. The Feed Area of the X-3346 building sublimes uranium hexafluoride (UF_6) for feed to the enrichment process and sublimes and desublimes UF_6 for blending/ transfer operations between cylinders and transfer of UF_6 material to customer cylinders for shipment as described in Section 1.1 of this license application and contains a variety of potential sources for radioactive effluents, both as gaseous UF_6 and particulate uranyl fluoride (UO_2F_2). These sources are vented to the atmosphere through an evacuation system located in the Withdrawal Area of this building, which has separate subsystems to control gaseous and airborne particulate effluents. Both sub-systems exhaust to a continuously monitored combined vent.

The Withdrawal Area of the X-3346 building withdraws and desublimes both the product and tail streams from the enrichment process as described in Section 1.1 of this license application and contains a variety of potential sources for radioactive effluents, both as gaseous UF_6 and particulate UO_2F_2. These sources are vented to atmosphere through an evacuation system located in the Withdrawal Area. There are separate evacuation systems, which have separate subsystems to control gaseous and airborne particulate effluents.

The cylinder burping/heeling system, feed ovens, freezer/sublimers, cold boxes, sampling system, and process piping in these areas are manifolded to the gaseous effluent side of their respective evacuation systems. Gases evacuated from process systems, which can contain high concentrations of UF_6, are processed through cold traps to desublime the UF_6 and separate it from the non-UF_6 gases. Residual gases leaving the cold trap have a very low concentration of UF_6, which is further reduced by passing the gas through an alumina trap. When an evacuation system cold trap becomes full, it is valved off from the vent and its contents sublimed to a dump cylinder so the material can be fed to the enrichment plant. The cold traps can be bypassed to allow rapid evacuation of a volume that does not contain radioactive material. The alumina traps cannot be bypassed.

Cylinder connections and disconnections have the greatest potential for small releases of UF_6 to the workspace. UF_6 released in this manner reacts quickly with ambient humidity to form UO_2F_2. A WISP system is used to collect these gases from fixed operational points (e.g. feed oven cylinder connection) through the evacuation system. Portable gulper systems are used to collect any small release of material during maintenance operations. Gulper systems utilize a flexible hose or hood to evacuate the air in the immediate area where the connection is being made or broken. The captured gases are passed through a roughing filter followed by a High Efficiency Particulate Air (HEPA) filter to collect the UO_2F_2 particulate. The portable gulpers are exhausted within the building in which they are being used.

The effluents from the WISP sub-systems are combined and vented to the atmosphere through a common vent after each subsystem associated with the evacuation system has removed the uranium. The vent is equipped with continuous gas flow monitoring instrumentation with local readout as well as the analytical instrumentation required to continuously sample, monitor and to alarm UF_6 breakthrough in the effluent gas stream. The continuous vent monitor/sampler is described in Section 9.2.2.1 of this chapter.

Ventilation air in the X-3346 is monitored under the Radiation Protection Program as described in Section 4.7 of this license application. Environmental Compliance personnel review summaries of the monitoring data at least quarterly to verify that ventilation exhausts are insignificant as defined in NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications, Revision 2* (SRP) (i.e., less than 3 x 10^{-13} microcuries per milliliter [μCi/mL] uranium).

<u>X-3001 and X-3002 Process Buildings</u>

The process buildings house the operating centrifuges that separate the UF_6 into enriched product and depleted tails as described in Section 1.1 of this license application and contain a limited variety of potential sources for radioactive effluents, primarily as gaseous UF_6. These sources are vented to atmosphere through either the Purge Vacuum (PV) or Evacuation Vacuum (EV) Systems. Both systems exhaust to a common continuously monitored vent.

Enrichment equipment operates at sub-atmospheric pressures. Equipment operation requires the removal of any air that leaks into the process. The PV/EV Systems are used to remove air in the enrichment equipment. Since the air may contain traces of UF_6 the gas removed by these systems is passed through a shared set of alumina traps prior to venting. The PV/EV systems in each half (north and south) of each process building are manifolded to one process building vent. For HALEU Demonstration, the PV/EV system is only in the north half (Train 3) of the X-3001 Process Building. Additionally, for HALEU, there is also a bank of Sodium Fluoride (NaF) traps to facilitate removal of UF_6 inventory from the cascade should it be necessary. The discharge of the NaF traps is subsequently routed to PV/EV systems. Each process building vent is equipped with continuous gas flow monitoring instrumentation with local readout, as well as analytical instrumentation to continuously sample, monitor, and alarm UF_6 breakthrough in the effluent gas stream. The continuous vent monitors/samplers are described in Section 9.2.2.1 of this chapter.

Ventilation air in the process buildings is monitored under the Radiation Protection Program as described in Section 4.7 of this license application. Environmental Compliance personnel review summaries of the monitoring data quarterly to verify that ventilation exhausts are insignificant as defined in the SRP (i.e., less than 3 x 10^{-13} μCi/mL uranium).

X-3344 Customer Services Building

The X-3344 operations are applicable to commercial ACP operations only and are not used in the HALEU Demonstration. The Customer Services Building liquefies UF_6 for quality control sampling of UF_6 material as described in Section 1.1 of this license application and also contains multiple potential sources for radioactive effluents, both as gaseous UF_6 and particulate UO_2F_2. These effluents are vented from X-3344 building through piping to an evacuation system in the X-3346 building through a continuously monitored combined vent.

The autoclaves, sampling manifolds, sample containers and piping and process piping are manifolded to the gaseous effluent side of the appropriate WISP evacuation system. Gases evacuated from process systems, which can contain high concentrations of UF_6, are processed through cold traps located in the X-3346 Withdrawal Area to desublime the UF_6 and separate it from the non-UF_6 gases. Residual gases leaving the cold trap have a very low concentration of UF_6, which is further reduced by passing the gas through an alumina trap. When an evacuation cold trap becomes full, it is valved off from the vent and its contents sublimed to a cylinder. The evacuation cold traps can also be bypassed to allow rapid evacuation of a volume that does not contain significant amounts of radioactive material. The alumina traps cannot be bypassed.

Cylinder connections and disconnections have the greatest potential for small releases of UF_6 to the workspace. UF_6 released in this manner reacts quickly with ambient humidity to form UO_2F_2. A WISP system is used to collect these gases from fixed operation points (e.g. feed oven cylinder connection) through the evacuation system. Portable gulper systems are used to collect any small release of material during maintenance operations. Gulper systems utilize a flexible hose or hood to evacuate the air in the immediate area where the connection is being made or broken. The captured gases are passed through a roughing filter followed by a HEPA filter to collect the UO_2F_2 particulate. The portable gulpers are exhausted within the building in which they are being used.

The effluents from the WISP sub-systems are combined and vented to the atmosphere through a common vent after each sub-system associated with the evacuation system in the X-3346 building has removed the uranium. The vent is equipped with continuous gas flow monitoring instrumentation with local readout as well as the analytical instrumentation required to continuously sample, monitor and to alarm UF_6 breakthrough in the effluent gas stream. The continuous vent monitor/sampler is described in Section 9.2.2.1 of this chapter.

Ventilation air in the X-3344 building is monitored under the Radiation Protection Program as described in Section 4.7 of this license application. Environmental Compliance personnel review summaries of the monitoring data at least quarterly to verify that ventilation exhausts are insignificant as defined in the SRP (i.e., less than 3×10^{-13} μCi/mL uranium).

X-3012 Process Support Building

The X-3012 building provides process control functions and maintenance support as described in Section 1.1 of this license application. The ACR provides central operating functions to monitor and control HALEU Demonstration processes in the X-3001 process building.

Ventilation air in the X-3012 building is monitored under the Radiation Protection Program as described in Section 4.7 of this license application. Environmental Compliance personnel review summaries of the monitoring data quarterly to verify that ventilation exhausts are insignificant as defined in the SRP (i.e., less than 3 x 10^{-13} µCi/mL uranium).

X-7725 Recycle/Assembly Building; X-7726 Centrifuge Training and Test Facility; and X-7727H Interplant Transfer Corridor

Centrifuges are assembled and may be disassembled for repair or inspection as described in Section 1.1 of this license application in either the X-7725 building or X-7726 facility. The extent to which a centrifuge is disassembled depends upon the nature of the fault. 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.

As described in Section 1.1 of this license application, some completely assembled centrifuges are tested with UF_6 in the Gas Test Stands in the commercial ACP operation. In the HALEU Demonstration, the X-7725 building will only be used for temporary storage and for interior transport to and from the X-7726 facility. This is a separate room within X-7725 building with its own ventilation and emission control system. UF_6 for the test stands is supplied from a small cylinder within this room. Exhaust from the test stands passes through alumina traps to a continuously monitored vent. The vent is equipped with continuous gas flow monitoring instrumentation with local readout, as well as the analytical instrumentation required to continuously sample, monitor, and to alarm UF_6 breakthrough in the effluent gas stream. The continuous vent monitor/sampler is described in Section 9.2.2.1 of this chapter.

Ventilation air in both the X-7725 building and X-7726 facility is monitored under the Radiation Protection Program as described in Section 4.7 of this license application. Environmental Compliance personnel review summaries of the monitoring data quarterly to verify that ventilation exhausts are insignificant as defined in SRP (i.e., less than 3 x 10^{-13} µCi/mL uranium).

As described in Section 1.1, the X-7727H corridor is used only to provide indoor transport for sealed components (e.g., individual centrifuges) between the X-7725 building and the process buildings and is closed off from these buildings except when such transport is actually occurring. Consequently, the X-7727H corridor is never directly exposed to a source of gaseous uranium although it does have some air transfer from the process buildings and X-7725 building. At worst, the airborne uranium concentration in the X-7727H corridor will not exceed that in the process buildings or X-7725 building. This is insignificant as defined in the SRP (i.e., less than 3 x 10^{-13} µCi/mL uranium).

Waste Management

The ACP obtains waste management services for various radiological and non-radiological materials. The radiological waste management services are obtained from a qualified provider licensed/certified by the NRC or an agreement state.

Laboratory Services

The ACP obtains analytical services for various radiological and non-radiological materials. The radiological analytical services are obtained from a qualified laboratory licensed/certified by the NRC or an agreement state.

9.2.1.2.2 Control of Liquid Effluents

The centrifuges and PV/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 exchangers to the Tower Water Cooling (TWC) system, which is cooled by a single cooling tower. Waste heat from the cold trap refrigeration systems in X-3346 and X-3356 buildings is also discharged to the TWC system. Currently, the TWC discharges its blowdown to the GDP Recirculating Cooling Water (RCW) system under a service agreement, which in turn discharges its blowdown directly to the Scioto River via an underground pipeline (National Pollutant Discharge Elimination System [NPDES] Outfall 004). The RCW system does not provide any treatment of the TWC blowdown; it simply provides a convenient pathway to a suitable permitted discharge point. At some point in the future, DOE is expected to decommission and decontaminate the GDP, including the RCW system. By that time, the TWC blowdown will have to be modified to bypass the RCW system and discharge directly to the RCW discharge pipeline. The schedule for this has not been established. There should be no licensed material in the TWC blowdown.

In the interim, the GDP RCW system has ample capacity to accept the TWC effluent without either physical modification or adjustment to its discharge limits The TWC system is currently fitted with three 10,800 gallon per minute pumps and even assuming a conservative blowdown rate of ten percent, TWC blowdown flow will be no more than 3,240 gallons per minute.

Discharges from the RCW System are monitored by an automated sampler, which collects a weekly composite sample of the liquid effluent for radiological analysis as well as sample(s) for NPDES-mandated analyses. This data is available to the ACP as assurance that no unanticipated discharge of licensed material has occurred.

Leakage from the MCW system and incidental spills of water elsewhere in the ACP, are 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 ACP. 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 analyzed prior to disposal. If the contents meet the requirements of 10 CFR 20.2003, they may be pumped to the reservation sanitary sewer system. Otherwise the tank contents will be containerized for off-site disposal. An integrity assurance plan developed by Engineering assures that the tanks are not leaking as the ACP take possession of them. This plan will be completed and will be added to this application as a reference prior to the NRC's pre-operational inspections. Following completion of this integrity assurance plan, inventory monitoring of the tank contents is used to detect leaks from the LEC System.

Storm water runoff from the ACP area, along with some once-through cooling water (sanitary water), drains to a pair of holding ponds.

- The X-2230N West Holding Pond (NPDES Outfall 012) provides a quiescent zone for settling suspended solids, dissipation of chlorine, and oil diversion and containment. The pond discharges to the same unnamed tributary of the Scioto River as X-230J-5. An automated sampler collects a weekly composite sample of the liquid effluent for radiological analysis as well as sample(s) for NPDES-mandated analyses.

- The X-2230M Southwest Holding Pond (NPDES Outfall 013) provides a quiescent zone for settling suspended solids, dissipation of chlorine, and oil diversion and containment. The pond discharges to an unnamed tributary of the Scioto River. An automated sampler collects a weekly composite sample of the liquid effluent for radiological analysis as well as sample(s) for NPDES-mandated analyses.

The X-6002 Recirculating Hot Water Plant, which provides heat to multiple buildings at the ACP, contains a particulate separator (NPDES Outfall 613) that removes suspended solids from the water used in the plant. Samples from the blowdown of the particulate separator are taken prior to its discharge to the DOE reservation sewage treatment plant (GDP NPDES Outfall 003).

Outdoor cylinder storage pads will be used in the commercial ACP operation; however, all cylinder storage will be maintained inside the X-3001 facility in the HALEU Demonstration. Most of the ACP cylinder storage pads are within the drainage of the X-2230M and X-2230N Holding Ponds. The ACP also uses cylinder storage pads on the north end of the reservation (X-745G-2 and X-745H). The ACP conducts an inspection and maintenance program for its UF_6 cylinders to ensure that no licensed material is released to the storage pads in accordance with USEC-651, *Uranium Hexafluoride: A Manual of Good Handling Practices*. Stormwater runoff from the north pads drains to holding ponds in accordance with a service agreement. Holding pond effluents are currently continuously monitored with automated samplers in accordance with the GDP environmental protection plan discussed in POEF-FBP-001, *Basis for Interim Operation of Former Uranium Enrichment Facilities (FUEF) at the Portsmouth Gaseous Diffusion Plant, Piketon, OH.* This data is available to ACP environmental personnel as assurance that no unanticipated discharge occurred.

9.2.1.3 As Low As Reasonably Achievable Reviews and Reports to Management

Action levels for control of both gaseous and liquid radioactive effluents from the ACP 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 required actions described in Table 9.2-1 include the analyses of trends in release data, evaluations of the probable impact of the releases and an assessment of the need for additional effluent controls to meet the ALARA goals. The Senior Shift Supervisor is responsible for assuring that action levels are acted upon.

The BEQs used in Table 9.2-1 is the maximum effluent expected under normal operation. BEQs have been established by the ACP environmental personnel and the responsible building management for every continuously monitored radiological vent and liquid discharge point to unrestricted areas. These BEQs are reviewed annually, at a minimum, by environmental personnel, the responsible building management and the ACP ALARA Committee to ensure the principles described in the ACP's ALARA policy are followed. This review also includes analyses of trends in radioactive effluents and environmental monitoring data. The results of this review are reported to the Production Support Manager and other senior management as described in Chapter 4.0 of this license application.

The specific values of the BEQs are listed in Table 9.2-2. The liquid release points are existing discharges and, while the ACP does not increase releases beyond historic levels, it does not decrease them either. Therefore, the liquid BEQs in Table 9.2-2 are based on GDP historic release rates.

9.2.1.4 Waste Minimization

Radioactive waste minimization and pollution prevention activities are coordinated by ACP environmental compliance and waste management personnel with the support of senior management.

Individual waste streams are identified and characterized based on process knowledge, routine radiation surveys as described in Chapter 4.0 and laboratory analysis, as needed. Generation of individual waste streams and waste management costs are tracked through a formal Request-for-Disposal database system administered by waste management personnel and the annual budgeting process.

Waste generating activities are evaluated for waste minimization opportunities with emphasis on those that generate hazardous wastes, low-level mixed wastes (LLMW), and low-level radioactive wastes (LLRW). Both LLMW and LLRW waste generation is inherently reduced in the ACP by the fact that the process operates under a high vacuum, which prevents radioactive material from escaping. Equipment that must be removed for maintenance is evacuated to the rest of the process first. The routine radiation surveys described in Chapter 4.0 of this license application verify that there is no spread of contamination within or out of the ACP. Hazardous waste generation is minimized by minimizing the procurement and use of hazardous substances. Waste that is generated is treated to the extent practical to reduce the volume, toxicity, or mobility before storage or disposal. The Licensee provides annual employee training that includes waste minimization information and encourages employee suggestions.

The Licensee provides environmental and waste management professionals with opportunities to attend offsite training and conferences for the purpose of seeking and exchanging technical information on waste minimization.

Waste minimization recommendations are evaluated by waste management and environmental compliance personnel and implemented, as appropriate, by waste management, materials procurement (for hazardous materials), and operations personnel.

This applies to ACP operations, associated support operations, and ACP subcontractors that generate waste.

9.2.2 Effluent and Environmental Monitoring

Based on historic GDP experience and operating plans, the radionuclides anticipated to be present in ACP gaseous effluents are ^{234}U, ^{235}U, and ^{238}U. The intention is to not introduce feedstock contaminated with significant concentrations of other nuclides into the process. Feed material that meets the American Standards for Testing and Materials (ASTM) specification for recycled feed may be used in the commercial ACP operation, which may contain radionuclides such as ^{236}U and Technetium (^{99}Tc). Feed material for the HALEU Demonstration could also be UF_6 meeting the ASTM UF_6 product standard, produced in former enrichment operations external to ACP (e.g. GDP operations). Based on historic GDP experience ^{99}Tc may eventually appear in some ACP gaseous effluents. The radionuclides anticipated to be present in ACP liquid effluents are ^{234}U, ^{235}U, ^{238}U, and ^{99}Tc, due to historic contamination of the reservation. Consequently, ACP effluents will be analyzed for these four nuclides as described in the applicable sections below.

9.2.2.1 Airborne Effluent Monitoring

9.2.2.1.1 Anticipated Effluent Levels

The maximum anticipated gaseous effluents from the ACP have been modeled using the EPA-approved and distributed dispersion model, CAP88-PC, and reservation meteorological data from calendar years 1998-2002. The results are summarized in Table 9.2-3. The maximum gaseous effluent anticipated under normal operations is 1.1 millicuries (mCi) of uranium over a week, or up to 0.057 curie (Ci) per year. The maximum exposed individual (MEI) for the ACP is located in the south-southwest sector of the reservation boundary. The projected maximum airborne concentration of total uranium due to ACP operations is only 3.2×10^{-15} µCi/mL, with an associated TEDE of 0.33 mrem. The uranium concentration is roughly three orders of magnitude lower than the applicable values in 10 CFR Part 20, Appendix B, Table 2. The projected TEDE due to ACP operations contributes roughly 66 percent to the ALARA goal given in Section 9.2.1.1 of this chapter, even assuming the average annual emission rates are equal to the maximum weekly emission rates. Average emission rates are expected to be much lower.

It is noted that HALEU Demonstration isotopic distributions may vary from these analyses performed for the commercial ACP operation, due to the use of enriched product as feed to the HALEU Demonstration. However, the HALEU Demonstration is limited to a cascade of only 16 centrifuges; whereas the original analyses for the commercial ACP operations were applicable to cascade containing thousands of centrifuges deployed in a cascade configuration with up to 3.8 million SWU/year. The commercial ACP analyses referenced in this section will conservatively bound any small variations in isotopic distribution that might be applicable to the HALEU Demonstration.

9.2.2.1.2 Demonstration of Compliance

Characterization of the radiological consequences of radionuclides released to the atmosphere from the ACP is accomplished by comparing measured emissions to the values in 10 CFR 20, Appendix B, Table 2 and the requirements of 10 CFR 20.1301, as applicable. The results are incorporated into semiannual reports submitted to the NRC in accordance with 10 CFR 70.59.

Characterization of the radiological consequences of radionuclides released to the atmosphere from the ACP is also 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 previous uranium enrichment operations at the reservation. The annual National Emission Standards for Hazardous Air Pollutants (NESHAP) Report includes the reservation identification, a description of plant operations (whether included under this license or not) during the previous year, the amount of radionuclides released to the atmosphere during the previous year, and the calculated TEDE to the most exposed member of the public.

Annual radionuclide releases to air are measured by the continuous vent samplers, as 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, the reported public dose includes gaseous radioactive effluents from the DOE reservation.

The dose calculations are made using either the original CAP88 package of computer codes or the CAP88-PC package distributed by the EPA. The CAP88/CAP88-PC packages contain an EPA approved version of the AIRDOS-EPA and DARTAB computer codes and the ALLRAD88 radionuclide data file. 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 are prepared from data collected from the reservation meteorological tower. Data from the National Weather Service may be used in lieu of or to supplement reservation meteorological data in the event the on-site tower becomes inoperable. The reservation 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.

Distances to the nearest residences are taken from U.S Geological Survey maps and population distributions are from the 2000 census data. EPA published default values for other off-site 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.

It is noted that HALEU Demonstration isotopic distributions may vary from these analyses performed for the commercial ACP operation, due to the use of enriched product as feed to the HALEU Demonstration. However, the HALEU Demonstration is limited to a cascade of only 16 centrifuges; whereas the original analyses for the commercial ACP operations were applicable to cascade containing thousands of centrifuges deployed in a cascade configuration with up to 3.8 million SWU/year. The commercial ACP analyses referenced in this section will conservatively bound any small variations in isotopic distribution that might be applicable to the HALEU Demonstration.

9.2.2.1.3 Monitoring of Gaseous Release Points

Each process vent in the X-3001, X-3002, X-3346, and X-7725 buildings has gas flow monitoring instrumentation with local readout as well as analytical instrumentation to continuously sample, monitor and to alarm UF_6 breakthrough in the effluent gas stream. The locations of these vents are shown in Figure 9.2-1. 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 sampler's electronic controller. The controller adjusts a control valve in the sample line to maintain a constant ratio between the vent and sample flows. The flow instruments are calibrated at least annually. The primary sample trap is equipped with an automated radiation monitor to continuously monitor the accumulation of uranium in the sampler. This radiation monitor provides the real-time indicator of effluent levels for operational control of the gaseous effluent control systems.

Detailed effluent calculations are based on laboratory analysis of the collected samples. Each vent sampler has two traps permanently dedicated to each trap position, with one in-service and the other either being processed or standing by to replace the in-service trap. Normally, 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 involved sampler traps are collected for immediate analysis as soon as the situation has stabilized. Alternatively, the sampling period may be extended, provided the sampler is operating continuously while the vent is operating. A hydrated alumina is used in the vent samplers to convert absorbed UF_6 to UO_2F_2. The UO_2F_2 does not easily separate from the alumina, so no special handling is necessary to avoid loss of uranium between sample collection and analysis. Annually, the sampler tubing and traps are also replaced and rinsed, and the rinsates analyzed for the same parameters as the alumina.

Vent samples are analyzed for ^{234}U, ^{235}U, ^{238}U, and ^{99}Tc as described in Section 9.2.2.5 of this chapter. GDP experience in uranium enrichment has shown that these three uranium isotopes account for more than 99 percent of the public dose due to uranium emissions. ^{99}Tc is a fission product that has contaminated much of the fuel cycle. Feed material that meets the ASTM specification for recycled feed may be used in the commercial ACP operation, which may contain additional radionuclides (i.e., ^{236}U and ^{99}Tc). Feed material for the HALEU Demonstration could also be UF_6 meeting the ASTM UF_6 product standard, produced in former enrichment operations external to ACP (e.g. GDP operations). Based on GDP historic experience ^{99}Tc may eventually appear in some ACP gaseous effluents. The ACP therefore monitors process vent samples for technetium as a precautionary measure.

Weekly gaseous effluents are calculated based on the primary trap analytical results and measured flows. These are compared to the action levels in Table 9.2-1 to determine whether gaseous effluents are threatening to exceed regulatory limits or ALARA goals. The weekly effluents are also accumulated to provide source terms for the annual public dose assessment required under 40 CFR Part 61. Quarterly and annual corrections to the accumulated weekly effluents are calculated based on the secondary trap and rinsate analyses, respectively, to complete the source terms.

Anticipated radionuclide concentrations in ventilation exhausts from occupied areas are insignificant as defined in the SRP. Radionuclide concentrations in room air are monitored as described in Section 4.7 of this license application. The results are reviewed by environmental engineers at least quarterly to verify that airborne concentrations are less than ten percent of the applicable values in 10 CFR Part 20, Appendix B, Table 2.

In the event of a radionuclide release outside the effluent monitoring system, the activity of the release will be estimated based on available data and engineering calculations (i.e., inventory data and mass balances).

9.2.2.1.4 Action Levels

Action levels for control of gaseous radioactive effluents from ACP 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 are the maximum effluents expected under normal operating conditions. BEQs have been established for every continuously monitored radiological vent. The specific BEQ values established for the monitored ACP vents are listed in Table 9.2-2.

9.2.2.1.5 Other Permits and Licenses

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 ACP therefore needs PTIs for its process vents. Within one year of the PTI being issued, the ACP also needs to apply to the Ohio EPA for a modification to its Title V permit to incorporate the entire ACP into the existing permit. The Title V permit supersedes the PTI once it is modified.

Sources of airborne radionuclides at DOE-owned plants are covered by an EPA Permit-By-Rule issued under 40 CFR Part 61, (NESHAP) Subpart H. This rule imposes a limit on airborne effluents of 10 mrem/year to the MEI, which applies to the entire reservation regardless of who "owns" any individual source within the reservation. The rule also requires an annual report, submitted by June 30 of the following year, detailing the processes at the reservation, the airborne effluents from each source, and annual TEDE to the MEI as calculated by a method approved by the EPA. A copy of this report is available to NRC as described in Section 9.3.2 of this chapter.

Also, under the NESHAP rule, new or modified sources of airborne radionuclides at DOE-owned plants 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. This will be necessary for the ACP since it has the potential to exceed this threshold.

9.2.2.2 Liquid Effluent Monitoring

9.2.2.2.1 Anticipated Effluent Levels

Anticipated routine radioactive effluents from the ACP are expected to be minimal. The bulk of liquid radioactive effluents from a uranium enrichment plant are decontamination and cleaning solutions. Centrifuges will not be routinely changed out, but routine maintenance such as instrument repair or repair to the PV/EV systems occurs. There are also maintenance activities that require cleaning and/or decontamination. The ACP uses dry decontamination methods to the extent practical to minimize liquid releases.

Spills are accumulated in the LEC system. The LEC collection tanks are sampled and analyzed for radioactive constituents prior to being emptied. If analysis indicates that LEC tank contents meet the criteria of 10 CFR 20.2003, the contents may be discharged to the reservation sanitary sewer. Otherwise, LEC tank contents will be containerized for disposal off-site. These are the only anticipated liquid discharges of licensed material from the ACP.

Actual sanitary wastewater (i.e., excluding LEC discharges) from the ACP is not anticipated to contain licensed radioactive material. Any licensed material that may be discharged will be released in accordance with the requirements of 10 CFR 20.2003. Consequently, anticipated radionuclide concentrations in the sanitary wastewater itself are anticipated to be insignificant as defined in the SRP.

There are no anticipated radioactive effluents from the MCW system, since it is a closed-loop system with no routine blowdown. The TWC system is a standard industrial recirculating water system with a routine blowdown stream to control the accumulation of solids within the cooling water. The TWC does not come in contact with licensed material unless there is leakage from the process to the MCW and then from the MCW to the TWC. This is unlikely since the MCW lines are on the outside of the centrifuge casings. Consequently, radionuclide concentrations in the TWC blowdown are also anticipated to be insignificant as defined in the SRP.

Storm water runoff and some once-through cooling water (sanitary water) flows through two holding ponds as described in Section 9.2.1.2.2 of this chapter, then discharges to the Scioto River in accordance with 10 CFR 20.1301. Radioactive materials in these streams are dominated either by naturally occurring radioactive materials or existing contamination from previous reservation operations. ACP effluents are not expected to cause any significant difference from historic release levels, which are insignificant as defined in the SRP.

The commercial ACP operation will use cylinder storage pads on the north end of the plant (X-745G-2 and X-745H). All cylinder storage will be maintained inside the X-3001 facility in the HALEU Demonstration. A cylinder inspection and maintenance program ensures that no licensed material is released to the storage pad. Nevertheless, runoff from the pads may drain to the existing X-230L North Holding Pond. This pond is maintained and monitored in accordance with 10 CFR 20.1301 and the monitoring data is available to the ACP. ACP operations are not expected to have any measurable impact on these ponds.

Anticipated radioactive releases from these points are summarized in Table 9.2-4, along with the limits from 10 CFR Part 20, Appendix B, Table 2 for comparison. The anticipated discharge levels are at least one order of magnitude below the Table 2 limits even before they mix with the Scioto River. Activity concentrations in the table are based on monthly grab samples from 1995 through 2000 for the X-2230M and X-2230N holding ponds. Activity concentrations for the other ACP-influenced continuous discharges are based on weekly composite samples from 1998 through 2002. Activity concentrations for the LEC system are based on the effluent being characterized prior to discharge.

No other ponds or impoundments at the ACP manage special nuclear material (SNM) and since the concentrations involved are well below the 10 CFR Part 20, Appendix B discharge limits, leakage to the soil is not a concern. The only underground tanks that potentially manage SNM are the LEC System described in Section 9.2.1.2.2 of this chapter. Inventory monitoring will be used to detect leakage from these tanks.

9.2.2.2.2 Demonstration of Compliance

Characterization of the radiological consequences of radionuclides released in liquid effluents from the ACP is accomplished by comparing measured concentrations to the values in 10 CFR Part 20, Appendix B, Tables 2 and 3 and the requirements of 10 CFR 20.1301 and 10 CFR 20.2003, as applicable. The results are incorporated into semiannual reports submitted to the NRC in accordance with 10 CFR 70.59.

Accumulated liquids in the LEC tanks are sampled for uranium and technetium prior to being removed from the tanks. ACP environmental personnel track the analytical results, volumes and disposition of the liquids. LEC liquids that do not meet the requirements of 10 CFR 20.2003 and 10 CFR Part 20, Appendix B, Table 3 are containerized for disposal at a suitable NRC-licensed site. LEC liquids that do meet the requirements of 10 CFR 20.2003 and 10 CFR Part 20, Appendix B, Table 3 may be either containerized for disposal off-site or discharged to the reservation sanitary sewer.

Sanitary wastewater from the ACP (exclusive of LEC effluents) is not expected to be contaminated with licensed material. Therefore, the ACP does not sample or analyze the untreated sewage. The sanitary sewer discharges to a sewage treatment plant located on the reservation that is regulated by both the DOE and the OEPA for radionuclides and which does sample and analyze it's effluent for uranium and technetium. This data is available to the ACP and is tracked by ACP environmental personnel against the applicable values 10 CFR Part 20, Appendix B, Table 2.

The other liquid effluent streams from the ACP are monitored as described in Section 9.2.2.2.3 of this chapter and compared to the applicable values in 10 CFR Part 20, Appendix B, Table 2 to demonstrate compliance with 10 CFR 20.1301. These streams are the TWC blowdown, X-2230M Southwest Holding Pond discharge, and X-2230N West Holding Pond discharge.

The ACP will use existing cylinder storage pads at the north end of the plant (X-745G-2 and X-745H). Runoff from the pads drain to the X-230J-5 Northwest Holding Pond and X-230L North Holding Pond, both of which are sampled and analyzed for uranium and technetium. This data is available to the ACP and these discharges will be tracked against the applicable values in 10 CFR Part 20, Appendix B, Table 2.

9.2.2.2.3 Monitoring of Liquid Release Points

There are only two ACP outfalls that discharge directly to publicly accessible areas, the X-2230M and X-2230N holding ponds. The locations of these outfalls are shown in Figure 9.2-2. The TWC blowdown discharges to a utility system (the RCW system) that provides a pathway to the Scioto River but does not provide any radiological treatment. These three discharges are equipped with automated samplers and continuous flow measurement. The flow monitors are calibrated at least annually. The combined discharge of the RCW system, the DOE reservation sewage treatment plant discharge and other reservation holding ponds are also equipped with automated samplers and continuous flow measurement. The data from these outfalls are available to the ACP as a defense in depth.

Outfall samples are analyzed for Gross Alpha and Gross Beta Activities, ^{99}Tc Activity and Total Uranium concentration as described in Section 9.2.2.5 of this chapter. Measurable Gross Alpha Activity is presumed to be due to uranium discharges from uranium enrichment operations, while Gross Alpha Activities below the Minimum Detectable Activity (MDA) are presumed to be due to naturally occurring radioactive materials. The isotopic distribution of enriched uranium discharges (i.e., ^{234}U, ^{235}U, and ^{238}U) is estimated to match the measured Gross Alpha Activity based on process knowledge. ^{99}Tc is a fission product that has contaminated much of the national fuel cycle and is present on the reservation. Measured ^{99}Tc concentrations in reservation outfalls have been falling for several years, but are detected occasionally. The ACP therefore routinely monitors radioactive effluents for technetium.

The only underground tanks in the ACP used to collect material that might contain radionuclides are the tanks of the LEC system. The LEC system consists of a set of drains and collection tanks primarily for collecting leaks and spills of chemically treated water. The drains are located throughout the process buildings. The tanks have a capacity of 550 gal each. Liquid level gauges mounted above grade on pipe stands monitor the tanks. Routine monitoring of the tanks' contents is based on observing and tracking the levels indicated on the gauges. Inventory tracking is relied on to indicate any leaks from the tanks. The contents of the LEC system will be sampled and analyzed for the same parameters as the continuous outfalls prior to disposal.

If analytical results indicate that LEC contents meet the requirements of 10 CFR 20.2003, they may be released to the reservation sanitary sewer system. Otherwise they will be containerized for disposal off-site.

9.2.2.2.4 Action Levels

Action levels for control of liquid radioactive effluents from the ACP 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 ACP liquid discharge point to unrestricted areas (i.e., X-2230M and X-2230N holding ponds) and for the TWC blowdown to the GDP area. BEQs have also been established for the LEC discharges, which are characterized before they are discharged, based on ten percent of the 10 CFR 20.2003 requirements. The specific BEQ values established for the ACP outfalls are listed in Table 9.2-2.

The ACP sanitary sewers, TWC blowdown, and runoff from the north cylinder storage pads discharge to DOE regulated units operated a service provider. The service provider has established and administers BEQ-based action levels for these discharges as documented in POEF-FBP-001, *Basis for Interim Operation of Former Uranium Enrichment Facilities (FUEF) at the Portsmouth Gaseous Diffusion Plant, Piketon, OH.*

9.2.2.2.5 Other Permits and Licenses

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 currently issued to the site, with two of them - covering all liquid discharges from the ACP. The third site NPDES permit is for the DUF_6 conversion facility. The ACP is required to submit a permit modification to collect all its discharge points into one or the other of the permits.

9.2.2.3 Waste Management

9.2.2.3.1 Waste Segregation and Collection

ACP generated wastes are collected and packaged by the individual(s) generating the waste. However, this is not appropriate in cases where waste would have to be "double handled" (e.g., surveying wastes expected to be contamination-free). In this case, it is most appropriate to survey prior to packaging. 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 developed in accordance with the commitments in Section 11.4 of this license application and 10 CFR Part 20.

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)
- Corrosives, Acids (Polybottles or polydrums)
- Scrap Metal (B-25 boxes or other similar boxes, and various drums)

In addition, 85- and 110-gal overpacks may be used for damaged containers if the wastes are appropriate for these size containers.

9.2.2.3.3 Radioactive Waste Storage

Those ACP wastes that are regulated for radiological content only are removed from the generating building and stored at an on-site radioactive waste storage area prior to final disposal. Those ACP wastes that are regulated for both radiological content and hazardous constituents and/or characteristics are stored at an on-site radioactive waste storage area under a conditional exemption for mixed waste (40 CFR Part 266, Subpart N [Federal] and Ohio Administrative Code-3745-266 [State]) prior to final disposal.

Other areas may be utilized as waste storage areas as required by plant operations. If outdoor storage is necessary, radioactive wastes with removable contamination are packaged in containers, and 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 on-site are stored until treatment is available at commercial treatment plants that are licensed in accordance with 10 CFR Part 61, or applicable NRC Agreement State requirements.

9.2.2.3.5 Off-site Waste Shipments

For Commercial ACP operation, off-site 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

ACP generated radioactive 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 Plan requirements.

9.2.2.3.7 Waste Tracking and Documentation

LLRW and LLMW generated at the ACP are 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, treatment data, and waste disposal information.

9.2.2.3.8 Other Permits and Licenses

The ACP is a generator of *Resource Conservation and Recovery Act* of 1976 hazardous wastes, which transfers solid wastes to appropriately permitted Treatment, Storage, and Disposal Facilities in accordance with applicable state and federal regulations.

9.2.2.4 Environmental Monitoring

The ACP is located contiguous to an existing uranium enrichment plant (the GDP) with approximately 50 years of accumulated experience in managing uranium and UF_6. The GDP was operated by the United States Enrichment Corporation, a subsidiary of USEC, from 1993 until it was placed in standby, and by predecessor organizations of the United States Enrichment Corporation prior to 1993. The environmental monitoring system for the ACP is based on the experience and data accumulated at the GDP.

9.2.2.4.1 Air Monitoring

Between 1980 and 1999, annual gaseous uranium effluents from the GDP ranged between 0.97 and 0.010 Ci/yr. Ambient air samples collected over this period by the GDP operators showed that these levels of effluents do not produce a quantifiable difference in ambient air concentrations in unrestricted areas. ACP operations are not expected to exceed these levels of effluents.

Consequently, ambient air monitoring is not useful in detecting or evaluating a public impact due to routine gaseous effluents from the ACP.

In addition, experience at the GDP has shown that any release large enough to produce high or intermediate consequences will first produce a large and very visible cloud of white smoke at the point of release. The ACP has a written procedure for dealing with unplanned releases ("See and Flee") that includes the immediate reporting of observed releases to the Senior Shift Supervisor (Operations Shift Supervisor during HALEU Demonstration)and evaluation by environmental professionals based on available credible information. Effluent monitoring will quantify routine gaseous effluents, but some accidental release scenarios may require information such as mass balances or measured environmental contamination to quantify an accidental release that did not pass through a monitored vent.

The United States Enrichment Corporation ceased sampling ambient air and returned the reservation's network of permanent air samplers to DOE in 1999, which upgraded the samplers for its own purposes. Based on the DOE Annual Environmental Reports published since 1999, average airborne uranium concentrations have been 1.1×10^{-15} micrograms per milliliter (μg/mL) on-site (i.e., within the DOE reservation), 7.4×10^{-16} μg/mL in unrestricted areas, and 5.5×10^{-16} μg/mL at the DOE background station. These results are consistent with the gross activity monitoring conducted prior to the turnover/upgrade. They are also a minimum of three orders of magnitude less than the applicable discharge limits for uranium isotopes in 10 CFR Part 20, Appendix B.

The reservation maintains a meteorological tower that is located on the southern section of the reservation. 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 reservation data.

The effluent monitoring and meteorological data are used to calculate the environmental impacts of airborne effluents from the ACP using EPA-approved dispersion models as described in Section 9.2.2.1 of this chapter.

9.2.2.4.2 Soil and Vegetation

Between 1980 and 2002, annual gaseous uranium effluents from the GDP have ranged between 0.97 and 0.005 Ci/yr. Soil and vegetation samples collected over this period by the GPD operators show that these levels of effluents do not produce a statistically significant difference in soil and vegetation concentrations in unrestricted areas. (Liquid effluents do not have a direct impact on soil and terrestrial vegetation around the reservation.) ACP operations are not expected to exceed these levels of effluents. Consequently, soil and vegetation monitoring is not useful in detecting a public impact due to gaseous effluents from the ACP. Therefore, atmospheric impacts of ACP operation, including action levels, will be based on gaseous effluent monitoring or other effluent information and atmospheric dispersion modeling as described in Section 9.2.2.1 of this chapter.

Soil and vegetation monitoring may be useful in assessing the long-term impacts of effluents from ACP operations or DOE environmental remediation projects or in assessing the impact of a high or intermediate consequence release that has already been detected and controlled. Therefore, the ACP maintains a soil and vegetation monitoring program for these purposes.

Soil and vegetation (wide-blade grass, typical of local cattle forage) samples are collected semiannually. The sampling networks completely surround the reservation, including the predominant downwind directions, and are administratively divided into on-site, off-site (up to 5 kilometers) and remote (5 to 16 kilometers off-site). A map of sampling locations in each group is provided in Figure 9.2-3. 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 chapter. See Table 9.2-5 for a summary of soil and vegetation results (1998-2002).

In addition to the semiannual vegetation samples, the ACP also 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 2002, annual waterborne uranium effluents from the GDP have ranged between 0.71 and 0.026 Ci/yr. Surface water samples collected over this period by the GDP operators show that these levels of effluents do not produce a statistically significant difference in the Scioto River. ACP operations are not expected to exceed these levels of effluents. Consequently, surface water monitoring is not useful in detecting or evaluating a public impact due to liquid effluents from the ACP. Therefore, impacts of ACP operation on local receiving waters, including action levels, will be based on effluent monitoring and pathways modeling as described in Section 9.2.2.2 of this chapter.

Surface water monitoring may be useful in assessing impacts of effluents from DOE environmental remediation projects or historical contamination. The ACP maintains a surface water monitoring program for this purpose.

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. A map of the sampling locations is found in Figure 9.2-4. Samples are collected weekly from the Scioto River and one location (RW8) in Little Beaver Creek. Other locations are sampled monthly. Specific details of the analytical methods are presented in Section 9.2.2.5 of this chapter. See Table 9.2-6 for a summary of surface water results (1998-2002).

9.2.2.4.4 Sediment Monitoring

Between 1980 and 2002, annual waterborne uranium effluents from the GDP have ranged between 0.71 and 0.026 Ci/yr. Sediment samples collected over this period by the GDP operators show that these levels of effluents do not produce a statistically significant difference in the Scioto River. ACP operations are not expected to exceed these levels of effluents. Consequently, sediment monitoring is not useful in detecting a public impact due to liquid effluents from the ACP. Therefore, impacts of ACP operation on local receiving waters, including action levels, will be based on effluent monitoring and pathways modeling as described in Section 9.2.2.2 of this chapter.

Sediment monitoring may be useful in assessing the long-term impacts of effluents from DOE environmental remediation projects or historical contamination. The ACP maintains a sediment monitoring program for this purpose.

Sediment sampling around the reservation is conducted semiannually to assess potential radionuclide accumulation in the surrounding receiving streams. The sampling locations include both upstream and downstream locations. A map of the sample locations is provided in Figure 9.2-5. 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 chapter. See Table 9.2-7 for a summary of sediment results (1998-2002).

9.2.2.4.5 Groundwater

Due to historical operations, the reservation has multiple plumes of groundwater contamination. The primary contaminate in the plumes is the halogenated solvent trichloroethylene, but limited areas of technetium contamination also exist.

DOE is conducting a site-wide environmental remediation program under an Agreed Order with the State of Ohio. As part of this program, reservation groundwater monitoring is under the control of DOE and the data is reported as part of DOE's Annual Environmental Report for the reservation. The ACP does not conduct a separate groundwater monitoring program. The current nuclides of interest in the DOE groundwater monitoring program are ^{99}Tc, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{240}Pu, and ^{241}Am.

9.2.2.4.6 Direct Gamma Radiation Monitoring

The only significant sources of environmental gamma radiation introduced to the reservation by man are the uranium isotope ^{235}U and the short-lived ^{238}U daughters. There are small amounts of other gamma emitters present on site as sealed sources and laboratory standards, but these are not detectable at any large distance. Gamma radiation levels in unrestricted areas around the ACP are dominated by naturally occurring radioactive materials.

The reservation conducts external gamma radiation monitoring consisting of lithium fluoride thermoluminescence dosimeters (TLDs) positioned at various site locations and at locations off-site. There are nine dosimeters spaced within Perimeter Road on the reservation; eight dosimeters spaced around the reservation boundary; and two dosimeters located off-site. Maps of the TLD locations are presented in Figures 9.2-6 and 9.2-7. 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 of the National Institute of Standards and Technology (NIST). See Table 9.2-8 for a summary of TLD results (1998-2002).

9.2.2.5 Laboratory Standards

A National Voluntary Laboratory Accreditation Program-certified service provider processes the site's environmental TLDs as described in Section 9.2.2.4.6. A laboratory licensed/certified by the NRC or an Agreement State provides other radiological and chemical analyses. The following description is based on services that have been provided by on-site laboratory in the past, but is not part of the ACP. Analytical laboratory services for the ACP will be conducted by certified provider to meet the equivalent standards as part of the contract.

Vent samples (i.e., activated alumina) 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 using recorded flow data and the measured weight of alumina in the sampler and to activity using published specific activities for individual isotopes. Gaseous effluents equivalent to an annual public dose of less than 0.1 mrem are routinely quantified. Since the airborne concentrations in 10 CFR Part 20, Appendix B, Table 2 are equivalent to an annual dose of 50 mrem, the MDA of these methods are equivalent to less than 0.2 percent of the 10 CFR Part 20, 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 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 6×10^{-5} µCi/mL for technetium. Consequently, the MDAs for liquid effluents are less than two percent of the applicable 10 CFR Part 20, Appendix B, Table 2 values.

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 system, formal written procedures, NIST-traceable standards, matrix spikes, duplicate, and replicate samples, check samples, and blind and double-blind QC samples.

The laboratories used shall participate in appropriate performance testing (PT) programs and maintain appropriate certifications for the types of analyses requested. For example, personnel safety monitoring analyses shall be performed by a laboratory certified by the American Industrial Hygiene Association for the analytes of interest, which would require them to successfully participate in PT programs for these analytes by performing them using National Institute of Occupational Safety and Health or Occupational Safety and Health Administration (OSHA) methodology.

Samples analyzed for environmental programs shall be performed by laboratories participating in appropriate certified PT programs, such as the following:

- EPA Discharge Monitoring Report-Quality Assurance Study for NPDES and Clean Water Act samples
- EPA Water Pollutant for waste water samples
- EPA Water Supply for drinking water samples.

9.2.2.6 Description of Status of Federal/State/Local Permits/Licenses

The ACP must comply with the applicable regulations under the *Atomic Energy Act* of 1954, as amended; 10 CFR Part 40; and 10 CFR Part 70 to hold a license to possess and use source and SNM. In addition, the ACP must comply with pertinent NRC regulations in 10 CFR Part 20 related to radiation dose limits to individual workers and members of the public. The Licensee is submitting an update to the previously approved Environmental Report to the NRC for the HALEU Demonstration program in accordance with 10 CFR Part 51.

As described in previous sections, the ACP will require PTIs from the State of Ohio to install all new air emission sources followed by a modification to the existing Title V air permit for the operation of those sources. The ACP will also be subject to the Radionuclide NESHAP administered by the EPA Region V. An additional PTI from the State of Ohio will be needed if the ACP installs any new wastewater lines. A modification to the existing NPDES permit will be needed to allow construction and operation of the ACP. These are the only Federal, State and local permits or other authorizations that the Licensee expects will be necessary for the ACP. Table 9.2-9 gives a full listing of the Federal, State and local permits and other authorizations and consultations that potentially could be required and the current status of each.

The ACP permit and reporting requirements will be incorporated and administered in the American Centrifuge Operating, LLC permits and reporting requirements until the Licensee establishes a compliance organization. The HALEU Demonstration, X-3001 purge vacuum and evacuation vacuum system, is currently incorporated in the American Centrifuge Operating, LLC Title V air permit (Permit Number P0115127).

Informal consultations have been made with the responsible agencies in compliance with the following:

- Section 7 of the *Endangered Species Act*
- *Fish and Wildlife Coordination Act*
- *National Historic Preservation Act* (NHPA), Section 106
- *Farmland Protection Policy Act* (FPPA)/Farmland Conservation Impact Rating

Consultation letters and responses are included in Appendix B of the accompanying Environmental Report.

9.2.3 Integrated Safety Analysis Summary

An Integrated Safety Analysis (ISA) Summary, meeting the requirements of 10 CFR 70.65(b), was prepared in accordance with the guidance contained in Chapter 3.0 of the SRP and NUREG-1513, *Integrated Safety Analysis Guidance Document.* The ISA Summary for the ACP is submitted for review (separate from this license application) as document LA-3605-0003, Integrated Safety Analysis Summary for the American Centrifuge Plant. Additionally, LA-3605-0003A, *Addendum 1 of the Integrated Safety Analysis Summary for the American Centrifuge Plant – HALEU Demonstration,* has also been developed and summarizes the ISA Summary for the HALEU Demonstration Program award by the DOE for the demonstration of the HALEU production to support DOE research and development activities and programs.

9.3 Reports to the Nuclear Regulatory Commission

9.3.1 10 *Code of Federal Regulations* 70.59 Reports

The ACP submits a written report to the NRC Regional Office and the Office of Nuclear Material Safety and Safeguards by March 1 and August 30 of the each year detailing: uranium and technetium (if any) amounts and concentrations in gaseous and liquid effluents during the previous reporting period (July through December and January through June, respectively) in accordance with 10 CFR 70.59. These reports also include an estimate of the public dose due to gaseous effluents over the previous year.

9.3.2 National Emission Standards for Hazardous Air Pollutants Reports

The ACP submits a written report to the EPA and OEPA by June 30 of each year detailing: plant operations and gaseous effluent monitoring during the previous calendar year, gaseous radioactive effluents over the previous year, an assessment of the public TEDE caused by those effluents, and an explicit comparison of the calculated TEDE to the EPA public dose limit (10 mrem annually). This report would become monthly if the maximum public TEDE exceeds 10 mrem annually.

This report is required under 40 CFR 61.94 and by the conditions of the Title V Permit issued by the State of Ohio. It also supports the requirement to demonstrate compliance with 10 CFR 20.1301 and 10 CFR 20.1101 as described in Section 9.2.2.1.2 of this chapter and is available upon request for inspection at the plant.

9.3.3 Baseline Effluent Quantity Reports

The ACP assesses any weekly effluent that exceeds any of the action levels as described in Table 9.2-1. Many years of experience by the GDP operators have shown that radioactive effluents less than the action levels in Table 9.2-1 cannot produce a public radiation dose that is within an order of magnitude of the dose restriction in 10 CFR 20.1101, let alone the dose limit of 10 CFR 20.1301. Any weekly effluent that exceeds the action levels in Table 9.2-1 requires a written estimate of the probable impact of the effluent, in conjunction with other monitored effluents from ACP operations, on the annual public radiation dose.

These reports are available on request by the NRC. They are not routinely submitted to outside authorities because they are considered interim assessments that are superseded by the semiannual reports and annual public dose assessment described in Sections 9.3.1 and 9.3.2 of this chapter.

In the event that evaluated releases threaten to exceed the public dose constraint in 10 CFR 20.1101, the NRC will be notified according to written procedures.

9.4 References

1. LA-3605-0002, *Environmental Report for the American Centrifuge Plant*
2. NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications*, Revision 2
3. U.S. Department of Energy, Portsmouth Annual Environmental Report for 2000, DOE/OR/11-3077&D1, December 2001
4. U.S. Department of Energy, Portsmouth Annual Environmental Report for 2001, DOE/OR/11-3106&D1, November 2002
5. 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
6. POEF-FBP-001, *Basis for Interim Operation of Former Uranium Enrichment Facilities (FUEF) at the Portsmouth Gaseous Diffusion Plant, Piketon, OH*
7. LA-3605-0003, *Integrated Safety Analysis Summary for the American Centrifuge Plant*

8. LA-3605-0003A, *Addendum 1 of the Integrated Safety Analysis Summary for the American Centrifuge Plant – HALEU Demonstration*

9. USEC-651, *Uranium Hexafluoride: A Manual of Good Handling Practices, Revision 9*

Table 9.2-1 American Centrifuge Plant Action Levels for Radionuclide Effluents

Weekly Sample Results		
Uranium [a]	**Technetium** [a]	**Required Actions** [b]
BEQ	BEQ	Review release data for previous six months for trends and estimate probable impact over calendar year. Evaluate whether additional 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 releases are ongoing or a single spike. Initiate investigation into cause(s) of increased releases. 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 RQ[c] (10 Ci in 24 hours)	Notify Operations Shift Supervisor [HALEU Demonstration operations] or Senior Shift Supervisor [commercial operations only] Trace source of abnormal releases and establish control or shutdown as needed. If releases cannot be mitigated within 24 hours, elevate to next level.
1 Ci[d]	8 Ci[d]	Close affected discharge points until control of releases is re-established.

[a] Uranium has an approximately 8-fold greater dose rate response than ^{99}Tc over air dominated exposure pathways. Uranium dose response completely dominates ^{99}Tc over water dominated exposure pathways.

[b] Required actions for any level include required actions listed under lower emission levels.

[c] RQ does not include permitted emissions. The ACP is regulated under 40 CFR Part 61, Subpart H for release of airborne radionuclides from the entire reservation up to the equivalent of 10 mrem/year TEDE to the most exposed member of the public.

[d] 1 Ci or 8 Ci in one weekly sample analysis.

Note: The Operations Shift Supervisor [HALEU Demonstration operations] or Senior Shift Supervisor [commercial operations only] has the authority to allow a restart.

Table 9.2-2 Baseline Effluent Quantities for American Centrifuge Plant Discharges

Release Point	Total Uranium	Technetium
Vents		
X-3001 North Vent	0.2 mCi/week	0.1 mCi/week [a]
X-3001 South Vent	0.2 mCi/week	0.1 mCi/week [a]
X-3002 North Vent	0.2 mCi/week	0.1 mCi/week [a]
X-3002 South Vent	0.2 mCi/week	0.1 mCi/week [a]
X-3346 Vent [b]	0.08 mCi/week	0.1 mCi/week [a]
X-7725 Gas Test Stands Vent	0.01 mCi/week	0.1 mCi/week [a]
Outfalls		
LEC Effluents [c]	3×10^{-7} µCi/ mL or 0.1 Ci/year	6×10^{-5} µCi/ mL or 0.1 Ci/year
X-2230N West Holding Pond (NPDES 012)	2.5×10^{-8} µCi/ mL	1.0×10^{-7}µCi/ mL
X-2230M Southwest Holding Pond (NPDES 013)	2.5×10^{-8} µCi/ mL	1.0×10^{-7}µCi/ mL
TWC System Blowdown	5.9×10^{-8} µCi/ mL	1.0×10^{-7}µCi/ mL

[a] Technetium BEQs for vents are based on five times the MDA.

[b] X-3346 Vent serves the X-3346 Feed and Withdrawal Areas and X-3344 Customer Services Building.

[c] LEC effluents are characterized before being discharged to the site sanitary sewer. The 100 mCi/yr standard includes uranium and technetium isotopes discharged to the site sanitary sewer during a calendar year.

Table 9.2-3 Anticipated Gaseous Effluents

Discharge Point	**Total Uranium** [a]		**Technetium**	
	μCi/ mL [b]	mCi/wk [c]	μCi/ mL [b]	mCi/wk [c]
X-3346 Feed and Withdrawal Building (1 vent)	$<3.2 \times 10^{-15}$	<0.08	1.2×10^{-16}	0
X-3001 and X-3002 Process Buildings (4 vents)		<0.8		0
X-7725 Gas Test Stands Vent		<0.01		0
XT-847 Glovebox Vent		0.0004		0.005
Laboratory Hoods [d]		0.17		0.035
10 CFR Part 20, App. B, Table 2	3×10^{-12}	------	8×10^{-9}	------

[a] Since uranium isotopes present at the ACP have the same discharge limit, uranium isotope activities are combined into a Total Uranium activity for simplify comparison to the Table 2 limits.

[b] Anticipated concentrations are maximum ambient concentrations at the DOE reservation boundary due to emission sources and are based on emission estimates and atmospheric dispersion modeling. Anticipated technetium concentration is based on no detectable releases from the X-7725 building and X-3000 series buildings.

[c] Anticipated discharges are measured at the vent and, by definition are less than the Baseline Effluent Quantities. Anticipated technetium discharges from the X-7725 building and X-3000 series buildings are zero.

[d] Bounding case for associated analytical services.

Table 9.2-4 Anticipated Liquid Effluents [a]

Discharge Point	Total Uranium [b] µCi/ mL	Technetium µCi/ mL
LEC Effluents	<3 x 10^{-7} and <0.1 Ci/yr	<2 x 10^{-8} (<MDA)
TWC System Blowdown	<3 x 10^{-8}	<2 x 10^{-8} (<MDA)
X-2230N West Holding Pond (NPDES Outfall 012) [e]	<1 x 10^{-8}	<2 x 10^{-8} (<MDA)
X-2230M Southwest Holding Pond (NPDES Outfall 013) [e]	<1 x 10^{-8}	<2 x 10^{-8} (<MDA)
Sanitary wastewater (excluding LEC effluents)	<3 x 10^{-8}	<2 x 10^{-8} (<MDA)
North Cylinder Pad Runoff	<1 x 10^{-8}	<2 x 10^{-8} (<MDA)
10 CFR Part 20, App. B, Table 2	3 x 10^{-7}	6 x 10^{-5}
10 CFR Part 20, App. B, Table 3	3 x 10^{-6}	6 x 10^{-4}

[a] ACP contributions only. Combined effluents from other site operations remain the responsibility of the individual operator.

[b] Since uranium isotopes present at the ACP have the same discharge limit, uranium isotope activities are combined into a Total Uranium activity to simplify comparison to the Table 2 limits.

[c] By definition, anticipated activity discharges are less than the BEQ.

[d] LEC effluents are characterized prior to discharge. One Ci/yr limit applies to combined uranium and technetium activities.

[e] Anticipated concentrations are annual averages based on monthly grab samples from 1995 through 2000.

Table 9.2-5 Environmental Baseline Activities/Concentrations 1998-2002

	Total Uranium µg/g	**Technetium** pCi/g	**Gross Alpha** pCi/g	**Gross Beta** pCi/g
Reservation (9 Sampling Locations) Soil				
Num. of Samples	117 (0)	117 (93)	117 (59)	117 (64)
Average	2.8	<0.2	<8	<14
Minimum	0.6	<0.1	<2	8
Maximum	4.4	1.5	21	36
Vegetation				
Num. of Samples	116 (113)	116 (103)	-----	-----
Average	<0.25	<0.3	-----	-----
Minimum	<0.04	<0.1	-----	-----
Maximum	0.9	7.3	-----	-----
Off Reservation (6 Sampling Locations) Soil				
Num. of Samples	74 (0)	74 (32)	74 (38)	74 (41)
Average	2.9	<0.2	<7	<14
Minimum	0.7	<0.1	<2	<8
Maximum	4.6	3.8	14	47
Vegetation				
Num. of Samples	73 (73)	73 (61)	-----	-----
Average	<0.24	<0.3	-----	-----
Minimum	<0.05	<0.1	-----	-----
Maximum	<0.34	3.3	-----	-----

The "number of samples" shows the total number of samples collected, including replicate and duplicate samples collected for quality assurance (QA) purposes, followed by the number of samples that were lower than the Minimum Detectable Concentration in parentheses. QA sample locations for soil and vegetation are assigned independently, so the number of samples in each group does not necessarily match.

Table 9.2-5 Environmental Baseline Activities/Concentrations 1998-2002

	Total Uranium µg/g	**Technetium** pCi/g	**Gross Alpha** pCi/g	**Gross Beta** pCi/g
Remote (12 Sampling Locations) Soil				
Num. of Samples	139 (0)	139 (133)	139 (73)	139 (77)
Average	3.0	<0.2	<7	<14
Minimum	0.7	<0.1	<3	<7
Maximum	5.9	0.8	16	22
Vegetation				
Num. of Samples	137 (80)	137 (128)	-----	-----
Average	<0.23	<0.2	-----	-----
Minimum	0.08	<0.1	-----	-----
Maximum	<0.28	<0.5	-----	-----
Background (4 Sampling Locations) Soil				
Num. of Samples	40 (0)	40 (36)	40 (17)	40 (26)
Average	3.5	<0.2	<8	<14
Minimum	1.7	<0.1	<5	<8
Maximum	6.8	0.5	16	25
Vegetation				
Num. of Samples	40 (23)	40 (37)	-----	-----
Average	<0.24	<0.2	-----	-----
Minimum	<0.14	<0.1	-----	-----
Maximum	0.28	0.5	-----	-----

The "number of samples" shows the total number of samples collected, including replicate and duplicate samples collected for QA purposes, followed by the number of samples that were lower than the Minimum Detectable Concentration in parentheses. QA sample locations for soil and vegetation are assigned independently, so the number of samples in each group does not necessarily match.

Table 9.2-6 Environmental Baseline Activities/Concentrations 1998 - 2002

	Total Uranium µg/L	**Technetium** pCi/L	**Gross Alpha** pCi/L	**Gross Beta** pCi/L
Surface Water/Upstream Big Run Creek				
Num. of Samples	60 (56)	60 (60)	60 (57)	60 (39)
Average	<1.3	<15	<5	<13
Minimum	<0.1	<6	<1	<6
Maximum	23.5	<28	<8	30
Surface Water/Downstream Big Run Creek				
Num. of Samples	118 (68)	118 (116)	118 (106)	118 (82)
Average	<1.5	<15	<6	<13
Minimum	0.2	<6	1	6
Maximum	23.2	<28	<140	33
Surface Water/Upstream Little Beaver Creek				
Num. of Samples	60 (59)	60 (60)	60 (56)	60 (41)
Average	<0.9	<15	<5	<11
Minimum	<0.1	<6	<1	<6
Maximum	1.3	<28	<12	<22
Surface Water/Downstream Little Beaver Creek				
Num. of Samples	321 (34)	322 (246)	322 (182)	322 (101)
Average	<1.7	<16	<6	<15
Minimum	<0.6	<8	2	<7
Maximum	9.4	43	44	78
Surface Water/Upstream Big Beaver Creek				
Num. of Samples	60 (36)	60 (58)	60 (48)	60 (25)
Average	<1.2	<16	<5	<14
Minimum	0.3	<8	2	<7
Maximum	5.8	<28	37	62

The "number of samples" shows the total number of samples collected, including replicate and duplicate samples collected for QA purposes, followed by the number of samples that were lower than the Minimum Detectable Concentration in parentheses.

Table 9.2-6 Environmental Baseline Activities/Concentrations 1998 - 2002

	Total Uranium µg/L	**Technetium** pCi/L	**Gross Alpha** pCi/L	**Gross Beta** pCi/L
Surface Water/Downstream Big Beaver Creek				
Num. of Samples	60 (50)	60 (58)	60 (51)	60 (36)
Average	<1.1	<16	<6	<14
Minimum	<0.1	<6	<1	<6
Maximum	5.2	<28	72	108
Surface Water/Upstream Scioto River				
Num. of Samples	261 (8)	261 (251)	261 (213)	261 (151)
Average	<1.9	<15	<6	<13
Minimum	<1.0	<6	2	<6
Maximum	32.6	<28	<13	40
Surface Water/Downstream Scioto River				
Num. of Samples	261 (6)	261 (254)	261 (206)	261 (156)
Average	<1.8	<16	<6	<13
Minimum	<1.0	<6	2	<7
Maximum	9.5	<29	86	34
Surface Water/Background Creeks				
Num. of Samples	240 (214)	240 (237)	240 (223)	240 (179)
Average	<1.0	<16	<4	<11
Minimum	<0.1	<6	<1	<6
Maximum	6.9	114[a]	11	46

The "number of samples" shows the total number of samples collected, including replicate and duplicate samples collected for QA purposes, followed by the number of samples that were lower than the Minimum Detectable Concentration in parentheses.

[a] One sample from a background location was analyzed at 114 picocuries per liter (pCi/L) of technetium, a beta emitter, but only 12 pCi/L of gross beta activity. The technetium activity is believed to be a case of cross contamination. The next highest technetium activity at the background locations was 28 pCi/L.

Table 9.2-7 Environmental Baseline Activities/Concentrations 1998 - 2002

	Total Uranium µg/g	Technetium pCi/g	Gross Alpha pCi/g	Gross Beta pCi/g
Sediment/X-2230M Southwest Holding Pond Discharge				
Num. of Samples	10 (0)	10 (6)	10 (4)	10 (4)
Average	3.8	<0.2	<9	<16
Minimum	1.8	<0.1	<4	<7
Maximum	6.2	0.3	18	<36
Sediment/X-2230N West Holding Pond Discharge				
Num. of Samples	13 (0)	13 (4)	13 (4)	13 (11)
Average	3.2	<0.3	<7	<11
Minimum	2.3	<0.1	<3	<7
Maximum	4.9	0.6	10	<17
Sediment/Upstream Little Beaver Creek				
Num. of Samples	15 (0)	15 (13)	15 (6)	15 (11)
Average	2.8	<0.1	<7	<13
Minimum	1.5	<0.1	<4	<7
Maximum	5.7	0.2	11	18
Sediment/X-230J-7 Discharge				
Num. of Samples	17 (0)	17 (0)	17 (7)	17 (4)
Average	5.9	7.1	<16	<32
Minimum	2.7	0.7	<5	<7
Maximum	21.2	31.3	83	170
Sediment/Downstream Little Beaver Creek				
Num. of Samples	28 (0)	28 (6)	28 (3)	28 (9)
Average	7.0	<64.5	<17	<85
Minimum	1.8	<0.1	<5	<10
Maximum	35.1	801[a]	61	924

The "number of samples" shows the total number of samples collected, including replicate and duplicate samples collected for QA purposes, followed by the number of samples that were lower than the Minimum Detectable Concentration in parentheses.

Table 9.2-7 Environmental Baseline Activities/Concentrations 1998 - 2002

	Total Uranium µg/g	**Technetium** pCi/g	**Gross Alpha** pCi/g	**Gross Beta** pCi/g
Sediment/Upstream Big Beaver Creek				
Num. of Samples	10 (0)	10 (2)	10 (4)	10 (6)
Average	2.1	<0.3	<7	<13
Minimum	0.9	<0.1	<5	<7
Maximum	4.6	0.7	9	25
Sediment/Downstream Big Beaver Creek				
Num. of Samples	10 (0)	10 (0)	10 (1)	10 (2)
Average	4.0	4.7	<11	<18
Minimum	2.8	1.1	<6	<12
Maximum	5.5	14.6	33	24
Sediment/Upstream Big Run Creek				
Num. of Samples	11 (0)	11 (8)	11 (3)	11 (8)
Average	3.8	<0.2	<7	<13
Minimum	2.3	<0.1	4	9
Maximum	4.8	<0.2	13	<17
Sediment/Downstream Big Run Creek				
Num. of Samples	29 (0)	29 (6)	29 (6)	29 (18)
Average	4.1	<0.8	<9	<14
Minimum	1.1	<0.1	<4	<7
Maximum	5.9	2.7	33	28
Sediment/Upstream Scioto River				
Num. of Samples	11 (0)	11 (11)	11 (7)	11 (8)
Average	2.1	<0.1	<7	<12
Minimum	0.9	<0.1	3	<7
Maximum	4.6	<0.2	<9	<17

The "number of samples" shows the total number of samples collected, including replicate and duplicate samples collected for QA purposes, followed by the number of samples that were lower than the Minimum Detectable Concentration in parentheses.

Table 9.2-7 Environmental Baseline Activities/Concentrations 1998 - 2002

	Total Uranium µg/g	Technetium pCi/g	Gross Alpha pCi/g	Gross Beta pCi/g
Sediment/Downstream Scioto River				
Num. of Samples	10 (0)	10 (8)	10 (5)	10 (6)
Average	2.1	<0.2	<9	<14
Minimum	1.4	<0.1	5	<8
Maximum	4.4	0.4	17	19
Sediment/Background Creeks				
Num. of Samples	40 (0)	40 (37)	40 (22)	40 (25)
Average	3.2	<0.2	<6	<13
Minimum	1.3	<0.1	<3	<7
Maximum	6.8	2.7	13	24

The "number of samples" shows the total number of samples collected, including replicate and duplicate samples collected for QA purposes, followed by the number of samples that were lower than the Minimum Detectable Concentration in parentheses.

[a] In Fall 2002, duplicate samples taken at the RM8 sample point contained 689 and 801 pCi/g of technetium. A replicate sample taken at the same time and a few yards away contained only 13 pCi/g of technetium. The RM8 sample taken the following spring contained only 13 pCi/g, which is consistent with previous samples.

Table 9.2-8 Environmental Baseline Radiation Levels 1998-2002

Area of Readings	Average	Minimum	Maximum
Reservation (includes 518, 737, 862, 906, 933, 1404A, A35, A36, and A40)	10.5 µRad/hr	6.4 µRad/hr	17.9 µRad/hr
X-746 Cylinder Yard (includes 874)	70.5 µRad/hr	60.1 µRad/hr	82.3 µRad/hr
Boundary (includes A3, A8, A9, A12, A15, A23, A24, and A29)	10.5 µRad/hr	6.2 µRad/hr	22.6 µRad/hr
Piketon (includes A6)	9.6 µRad/hr	7.4 µRad/hr	13.9 µRad/hr
Camp Creek (includes A28)	10.4 µRad/hr	7.8 µRad/hr	14.9 µRad/hr

Note: Locations ACP-1, ACP-2, ACP-3, ACP-4, and ACP-5 are new monitoring locations that will be established as the ACP is built.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Air Quality Protection			
Title V Operating Permit: Required for sources that are not exempt and are major sources, affected sources subject to the Acid Rain Program, sources subject to new source performance standards (NSPS), or sources subject to National Emission Standards for Hazardous Air Pollutants (NESHAPs).	Ohio Environmental Protection Agency (OEPA); U.S. Environmental Protection Agency (EPA)	*Clean Air Act* (CAA), Title V, Sections 501-507 (*U.S. Code*, Title 42, Sections 7661-7661f [42 USC 7661-7661f]); *Ohio Administrative Code* (OAC) 3745-77-02	Centrus Energy Corp. (the Licensee) is the holder of a final Title V Operating Permit (Facility ID 0666000000) with an issue date of July 27, 2017 and effective date of August 17, 2017. The plant is subject to *Code of Federal Regulations*, Title 40, Part 61, Subpart H (40 CFR Part 61, Subpart H), "National Emissions Standards for Emissions of Radionuclides which is included in the terms and conditions of the Title V Operating Permit.
Ohio Permit to Install (PTI): Required for (1) any source to which one or more of the following CAA programs would apply: prevention of significant deterioration (PSD), nonattainment area, NSPS, and/or NESHAPs; and (2) any source to which one or more of the following state air quality programs would apply; Gasoline Dispensing Facility Permit, Direct Final Permit, and/or Small Maximum Uncontrolled Emissions Unit Registration.	OEPA	CAA, Title I, Sections 160-169 (42 USC 7470-7479); OAC 3745-31-02	The Licensee has determined that the PSD, nonattainment area, and NSPS programs do not apply to the ACP. However, air emission sources requiring an Ohio PTI would apply to the ACP and the Licensee will submit a timely PTI application to the OEPA.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Air Quality Protection (Cont.)			
Ohio Permit to Operate: Required for (1) any source to which one or more of the following CAA programs would apply; PSD, nonattainment area, NSPS, NESHAPs; and (2) any source to which one or more of the following state air quality programs would apply: State Permit to Operate and/or registration of operating unit with potential air emissions of an amount and type considered minimal; this permit is not required, however, for any facility that must obtain a Title V Operating Permit.	OEPA	CAA, Title I, Sections 160-169 (42 USC 7470-7479); OAC 3745-35-02	Centrus Energy Corp. (the Licensee) is the holder of a final Title V Operating Permit (Facility ID 0666000000) with an issue date of July 27, 2017 and effective date of August 17, 2017Sources requiring a PTI will be incorporated in the Title V Operating Permit.
Risk Management Plan (RMP): Required for any stationary source that has regulated substance (e.g., chlorine, hydrogen fluoride, nitric acid) in any process (including storage) in a quantity that is over the threshold level.	EPA; OEPA	CAA, Title 1, Section 112(r) (7) (42 USC 7412); 40 CFR Part 68; OAC 3745-104	The Licensee has determined that no regulated substances would be stored at the ACP in quantities that exceed the threshold levels. Accordingly, an RMP will not be required.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Air Quality Protection (Cont.)			
CAA Conformity Determination: Required for each criteria pollutant (i.e., sulfur dioxide, particulate matter, carbon monoxide, ozone, nitrogen dioxide, and lead) where the total of direct and indirect emissions in a nonattainment or maintenance area caused by a federal action would equal or exceed threshold rates.	OEPA	CAA, Title 1, Section 176 (c) (42 USC 7506); 40 CFR 93; OAC 3745-102;	Pike County, Ohio has been designated as "Cannot be Classified or Better Than Standard" for criteria pollutants. Because the county is in attainment with National Ambient Air Quality Standards for criteria pollutants and contains no maintenance areas, no CAA conformity determination is required for any criteria pollutant that would be emitted as a result of the proposed action. Existing air quality on the site is in attainment with National Ambient Air Quality Standards (NAAQS) for the criteria pollutants.
Water Resources Protection			
National Pollutant Discharge Elimination System (NPDES) Permit – Construction Site Storm Water: Required before making point source discharges into waters of the state of storm water from a construction project that disturbs more than 5 acres (2 ha) of land.	OEPA	*Clean Water Act* (CWA) (33 USC 1251 et seq.); 40 CFR Part 122; OAC-3745-33-02, 3745-38-02, and 3745-38-06	The Licensee has determined that construction of the ACP and new cylinder storage yards would require an NPDES Permit for the construction site storm water discharges. Centrus Energy Corp. is the holder of NPDES Permit number 0IS00023ED. If requested, a Storm Water Pollution Prevention Plan (SWPPP) will be submitted to the OEPA at the appropriate time. Storm water will discharge through existing outfalls covered by a NPDES Permit.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Water Resources Protection (Cont.)			
National Pollutant Discharge Elimination System (NPDES) Permit – Industrial Facility Storm Water: Required before making point source discharges into waters of the state of storm water from an industrial site.	OEPA	CWA (33 USC 1251 et seq.); 40 CFR Part 122; OAC-3745-33-02, 3745-38-02, and 3745-38-06	The Licensee has determined that storm water would be discharged from the ACP site during operations. Storm water will discharge through existing outfalls covered by a NPDES Permit.
National Pollutant Discharge Elimination System (NPDES) Permit – Process Water Discharge: Required before making point source discharges into waters of the state of industrial process wastewater.	OEPA	CWA (33 USC 1251 et seq.); 40 CFR Part 122; OAC-3745-33-02, 3745-38-02, and 3745-38-06	The ACP will process industrial wastewater through an existing NPDES permitted facility and through existing outfalls covered by the NPDES Permit.
Ohio Surface Water PTI: Required before constructing sewers or pump stations.	OEPA	OAC-3745-31-02	If required, before construction of sewer lines and pump stations at the ACP a PTI to modify the existing NPDES permit would be submitted to the OEPA at the appropriate time.
Ohio Surface Water PTI: Required before constructing any wastewater treatment or collection system or disposal facility.	OEPA	OAC-3745-31-02	If required, a PTI to modify the existing NPDES permit would be submitted to the OEPA at the appropriate time.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Water Resources Protection (Cont.)			
CWA Section 404 (Dredge and Fill) Permit: Required to place dredged or fill material into waters of the United States, including areas designated as wetlands, unless such placement is exempt or authorized by a nationwide permit or a regional permit; a notice must be filed if a nationwide or regional permit applies.	U.S. Army Corps of Engineers (USACE)	CWA (33 USC 1251 et seq.); 33 CFR Parts 323 and 330	The Licensee believes that construction of the ACP would not result in dredging or placement of fill material into wetlands within the jurisdiction of the USACE. If construction activities are subject to the CWA Section 404 Permit program, they may be covered under a USACE Nationwide CWA Section 404 Permit (i.e., No. 14 [Linear Transportation Projects], 18 [Minor Discharges], or 19 [Minor Dredging]). If necessary, the Licensee will consult with the USACE concerning the project and, if appropriate, submit either a pre-construction notification about activities covered by a nationwide permit or an application for an individual Section 404 Permit.
Ohio General Permit for Filling Category 1 and Category 2 Isolated Wetlands: Required where the proposed project involves the filling or discharge of dredged material into Category 1 and Category 2 isolated wetlands, causing impacts that total 0.5 acre (0.20 ha) or less.	OEPA	*Ohio Revised Code* (ORC) Sections 6111.021-6111.029	The Licensee believes that construction of the ACP would not result in dredging or placement of fill material into wetlands within the jurisdiction of the OEPA isolated wetlands program. However, if necessary, submit to the OEPA a Pre-Activity Notice of activities covered under the General Permit for Filling Isolated Wetlands.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Water Resources Protection (Cont.)			
Ohio Individual Isolated Wetland Permit: Required where the proposed project involves the filling or discharge of dredged material into Category 1 and Category 2 isolated wetlands, causing impacts that total greater than 0.5 acre (0.20 ha) for Category 1 isolated wetlands and/or greater than 0.5 acre (0.20 ha) but not exceeding 3 acres (1.21 ha) for Category 2 isolated wetlands.	OEPA	ORC Sections 6111.021-6111.029	The Licensee believes that construction of the ACP would not result in dredging or placement of fill material into wetlands within the jurisdiction of the OEPA isolated wetlands program. Accordingly, the Licensee will consult, if necessary, with the OEPA concerning the project and, if appropriate, submit to the OEPA an application for an Individual Isolated Wetland Permit.
Spill Prevention Control and Countermeasures (SPCC) Plan: Required for any facility that could discharge oil in harmful quantities into navigable waters or onto adjoining shorelines.	EPA	CWA (33 USC 1251 et seq.); 40 CFR Part 112	SPCC plan ESH-343-09-018 has been developed and approved for the American Centrifuge Plant.
CWA Section 401 Water Quality Certification: Required to be submitted to the agency responsible for issuing any federal license or permit to conduct an activity that may result in a discharge of pollutants into waters of a state.	OEPA	CWA, Section 401 (33 USC 1341); ORC Chapters 119 and 6111; OAC Chapters 3745-1, 3745-32, and 3745-47	The Licensee believes that it would not be required to obtain a CWA Section 401 Water Quality Certification for construction or operation of the ACP or new cylinder storage yards. If the Licensee determines that a federal license or permit is required (e.g., a CWA Section 404 Permit), a CWA Section 401 Water Quality Certification will be requested from the OEPA at the appropriate time.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Water Resources Protection (Cont.)			
Public Water System: A completed application for an initial public water system license is required prior to the operation of the public water system.	OEPA	OAC-3745-84-01(B)(b)	The Licensee will procure services from a qualified vendor.
Underground Storage Tank (UST) Installation Permit: Required before beginning installation of a UST system (i.e., a tank and/or piping of which 10 percent or more of the volume is underground and that contains petroleum products or substances defined as hazardous by the Comprehensive Environmental Response, Compensation, and Liability Act [CERCLA], except those hazardous substances that are also defined as hazardous waste by the RCRA).	Ohio Department of Commerce, Ohio Bureau of Underground Storage Tank Regulations (BUSTR)	OAC 1301:7-9-06(D)	One UST system is installed at the ACP. Registration number: 66005107-R00010 Tank Number: T00016
New UST System Registration: Required within 30 days of bringing a new UST system into service.	EPA; Ohio BUSTR	RCRA, as amended, Subtitle I (42 USC 6991a-6991i); 40 CFR 280.22; OAC 1301:7-9-04	If new UST systems would be installed at the ACP the Registration would be filed at the appropriate time.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Water Resources Protection (Cont.)			
Above Ground Storage Tank (AST): A PTI required to install, remove, repair or alter any stationary tank for the storage of flammable or combustible liquids.	Ohio Department of Commerce, State Fire Marshal	OAC 1301:7-7-28(A)(3) 40 CFR 112.8	AST fuel storage tanks will be required for the ACP. Permits to install will be filed at the appropriate time.
Waste Management and Pollution Prevention			
Submit Determination Results: Required when a person who generates waste in the State of Ohio or a person who generates waste outside the state that is managed inside the state determines that the waste he/she generates is hazardous waste.	OEPA	OAC 3745-52-11	Upon characterization of newly generated waste streams from the ACP, notification would be made to the OEPA.
Registration and Hazardous Waste Generator Identification Number: Required before a person who generates over 220 lb (100 kg) per calendar month of hazardous waste ships the hazardous waste off-site.	EPA; OEPA	*Resource Conservation and Recovery Act* (RCRA), as amended (42 USC 6901 et seq.), Subtitle C; OAC 3745-52-12	Centrus Energy Corp. Hazardous Waste Generator Identification Number OHD987054723.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Waste Management and Pollution Prevention (Cont.)			
Construction and Demolition Debris Facility License: Required before establishing, modifying, operating, or maintaining a facility to dispose of debris from the alteration, construction, destruction, or repair of a man-made physical structure; however, the debris to be disposed of must not qualify as solid or hazardous waste; also, no license is required if debris from site clearing is used as fill material on the same site.	OEPA or Pike County Board of Health	OAC 3745-37-01	Construction debris would not be disposed of on site at the ACP. Therefore, no Construction and Demolition Debris Facility License would be required.
Low-Level Radioactive Waste Generator Report: Required within 60 days of commencing the generation of low-level waste in Ohio.	Ohio Department of Health	OAC 3701:1-54-02	The Licensee will file a Low-Level Radioactive Waste Generator Report with the Ohio Department of Health at the appropriate time. ODH ID Number 52-2107911

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Waste Management and Pollution Prevention (Cont.)			
Hazardous Waste Facility Permit: Required if hazardous waste will undergo nonexempt treatment by the generator, be stored on site for longer than 90 days by the generator of 2,205 lb (1,000 kg) or more of hazardous waste per month, be stored on site for longer than 180 days by the generator of between 220 and 2,205 lb (100 and 1,000 kg) of hazardous waste per month, disposed of on site, or be received from off-site for treatment or disposal.	EPA; OEPA	RCRA, as amended (42 USC 6901 et seq.), Subtitle C; OAC 3745-50-40	Hazardous waste would not be disposed of on site at the ACP. Should ACP become a large quantity generator and waste require storage on site for greater than 90 days for characterization, profiling, or scheduling for treatment or disposal a Hazardous Waste Facility Permit would be required and submitted at the appropriate time.
Low-Level Mixed Waste (LLMW): LLMW is a waste that contains both low-level radioactive waste and RCRA hazardous waste.	OEPA	OAC 3745-266; 40 CFR Part 266 Subpart N	The Licensee will manage LLMW in compliance with 40 CFR Part 266 Subpart N and Ohio Administrative Code Chapter 3745-266.
Industrial Solid Waste Landfill Permit to Install: Required before constructing or expanding a solid waste landfill facility in Ohio.	OEPA	OAC 3745-29-06	Industrial solid waste would not be disposed of on site at the ACP. Therefore, no Industrial Solid Waste Landfill Permit to Install would be required.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Emergency Planning and Response			
List of Material Safety Data Sheets: Submission of a list of material Safety Data Sheets is required for hazardous chemicals (as defined in 29 CFR Part 1910) that are stored on site in excess of their threshold quantities.	Local Emergency Planning Commission (LEPC); Ohio State Emergency Response Commission (SERC)	*Emergency Planning and Community Right-to-Know Act* of 1986 (EPCRA), Section 311 (42 USC 11021); 40 CFR 370.20; OAC 3750-30-15	The Licensee will prepare and submit a List of Material Safety Data Sheets at the appropriate time.
Annual Hazardous Chemical Inventory Report: Submission of the report is required when hazardous chemicals have been stored at a facility during the preceding year in amounts that exceed threshold quantities.	LEPC; Ohio SERC; local fire department	EPCRA, Section 312 (42 USC 11022); 40 CFR 370.25; OAC 3750-30-01	The Licensee will prepare and submit an Annual Hazardous Chemical Inventory Report each year. Centrus Energy Corp. Facility ID Number 45661NTDST3930U

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Emergency Planning and Response (Cont.)			
Notification of On-Site Storage of an Extremely Hazardous Substance: Submission of the notification is required within 60 days after on-site storage begins of an extremely hazardous substance in a quantity greater than the threshold planning quantity.	Ohio SERC	EPCRA, Section 304 (42 USC 11004); 40 CFR 355.30; OAC 3750-20-05	The Licensee will prepare and submit the Notification of On-Site Storage of an Extremely Hazardous Substance at the appropriate time, if such substances are determined to be stored in a quantity greater than the threshold planning quantity at the ACP. Facility ID Number 45661NTDST3930U
Annual Toxic Release Inventory (TRI) Report: Required for facilities that have 10 or more full-time employees and are assigned certain Standard Industrial Classification (SIC) codes.	EPA:OEPA	EPCRA, Section 313 (42 USC 11023); 40 CFR Part 372; OAC 3745-100-07	The Licensee will prepare and submit a TRI Report to the EPA as appropriate. Facility ID Number 45661NTDST3930U.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Emergency Planning and Response (Cont.)			
Transportation of Radioactive Wastes and Conversion Products Certificate of Registration: Required to authorize the registrant to transport hazardous material or cause a hazardous material to be transported or shipped.	U.S. Department of Transportation (DOT)	*Hazardous Materials Transportation Act* (HMTA), as amended by the *Hazardous Materials Transportation Uniform Safety Act* of 1990 and other acts (49 USC 1501 et seq.); 49 CFR 107.608(b)	Centrus Energy Corp. Certificate of Registration Number 052803005022LN.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Emergency Planning and Response (Cont.)			
Transportation of Radioactive Wastes and Conversion Products Packaging, Labeling, and Routing Requirements for Radioactive Materials: Required for packages containing radioactive materials that will be shipped by truck or rail.	DOT	HMTA (49 USC 1501 et seq.); *Atomic Energy Act* (AEA), as amended (42 USC 2011 et seq.); 49 CFR Parts 172, 173, 174, 177, and 397	When shipments of radioactive materials are made, the Licensee will comply with DOT packaging, labeling, and routing requirements.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Other			
Land Resources			
Farmland Protection and Policy Act (FPPA): Prime farmland is land that has the best combination of physical and chemical characteristics for producing crops of statewide or local importance. Prime farmland is protected by the Farmland Protection and Policy Act (FPPA) of 1981 which seeks "… to minimize the extent to which federal programs contribute to the unnecessary and irreversible conversion of farmlands to nonagricultural uses…"	U.S. Department of Agriculture	Farmland Protection and Policy Act (FPPA) of 1981 Public Law 97-98; 7 USC 4201[b]; 7 CFR Part 7, paragraph 658	Consultation letters are included in Appendix B LA-3605-0002, Environmental Report for the American Centrifuge Plant.
Biotic Resources			
Threatened and Endangered Species Consultation: Required between the responsible federal agencies and affected states to ensure that the project is not likely to (1) jeopardize the continued existence of any species listed at the federal or state level as endangered or threatened or (2) result in destruction of critical habitat of such species.	U.S. fish and Wildlife Service; Ohio Department of Natural Resources	*Endangered Species Act* of 1973, as amended (16 USC 1531 et seq.); ORC 1531.25-26 and 1531.99	Consultation letters are included in Appendix B LA-3605-0002, Environmental Report for the American Centrifuge Plant.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Cultural Resources			
Archaeological and Historical Resources Consultation: Required before a federal agency approves a project in an area where archaeological or historic resources might be located.	Ohio State Historic Preservation Officer (SHPO)	*National Historic Preservation Act* of 1966, as amended (16 USC 470 et seq.); *Archaeological and Historical Preservation Act* of 1974 (16 USC 469-469c-2); *Antiquities Act* of 1906 (16 USC 431 et seq.); *Archaeological Resources Protection Act* of 1979, as amended (16 USC 470aa-mm)	The Licensee has consulted with the Ohio SHPO regarding previous archeological and architectural surveys at the DOE reservation. Consultation letters are included in Appendix B of LA-3605-0002, *Environmental Report for the American Centrifuge Plant.*

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Other (cont.)			
Environmental Report (ER) Required by 10 CFR Part 51, this ER is being submitted to the U.S. Nuclear Regulatory Commission (NRC) to support licensing of the ACP.	NRC	*National Environmental Policy Act* of 1969, as amended (NEPA) (42 USC 4321 et seq.); 40 CFR Parts 1500-1508; 10 CFR Part 1021; 10 CFR Part 51 P.L. 91-190	LA-3605-0002, *Environmental Report for the American Centrifuge Plant* was prepared in accordance with the *U.S. Code of Federal Regulations*, 10 CFR Part 51, which implements the requirements of the National Environmental Policy Act (NEPA) of 1968, as amended (P.L.91-190).
Depleted UF_6 Management Measures: Establishes requirements for management, inspection, testing, and maintenance associated with the ACP Depleted UF_6 storage yards and cylinders owned by the Licensee at the DOE reservation as stipulated in the ACP License Application.	OEPA	OAC 3745-266; 40 CFR Part 266 Subpart N	The Licensee will manage the ACP Depleted UF_6 tails cylinders in accordance with 40 CFR Part 266 Subpart N and Ohio Administrative Code Chapter 3745-266 while in storage.

Table 9.2-9 Potentially Applicable Consents for the Construction and Operation of the American Centrifuge Plant

License, Permit, or Other Consent	Responsible Agency	Authority	Relevance and Status
Other (Cont.)			
Standard Industrial Classification (SIC): The SIC system serves as the structure for collection, aggregation, presentation, and analysis of the U.S. economy. An industry consists of a group of establishments primarily engaged in producing or handling the same product or group of products or in rendering the same services.	OSHA	SIC system	SIC 2819 Industrial Inorganic Chemicals, Not Elsewhere Classified

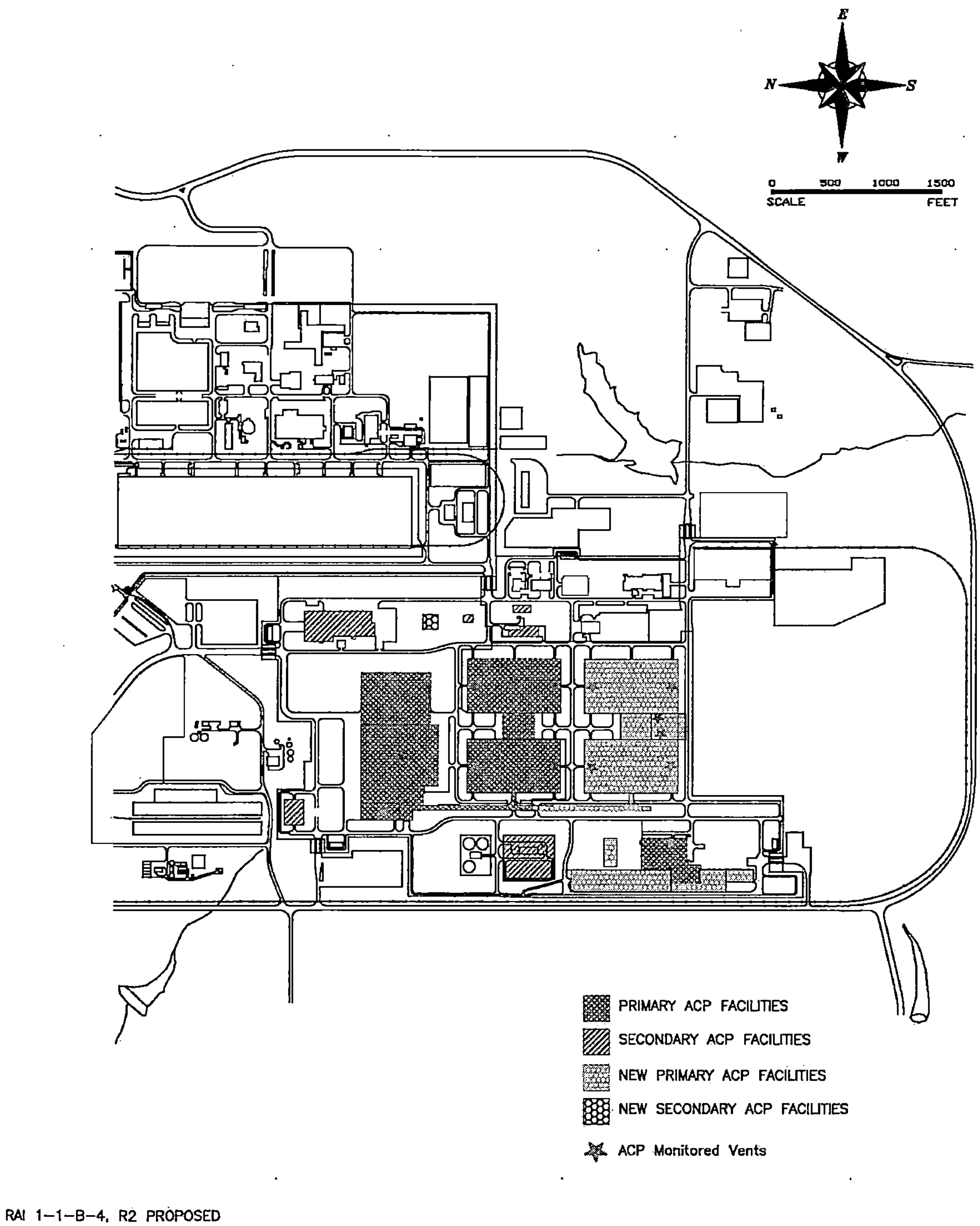

Figure 9.2-1 Locations of American Centrifuge Plant Monitored Vents

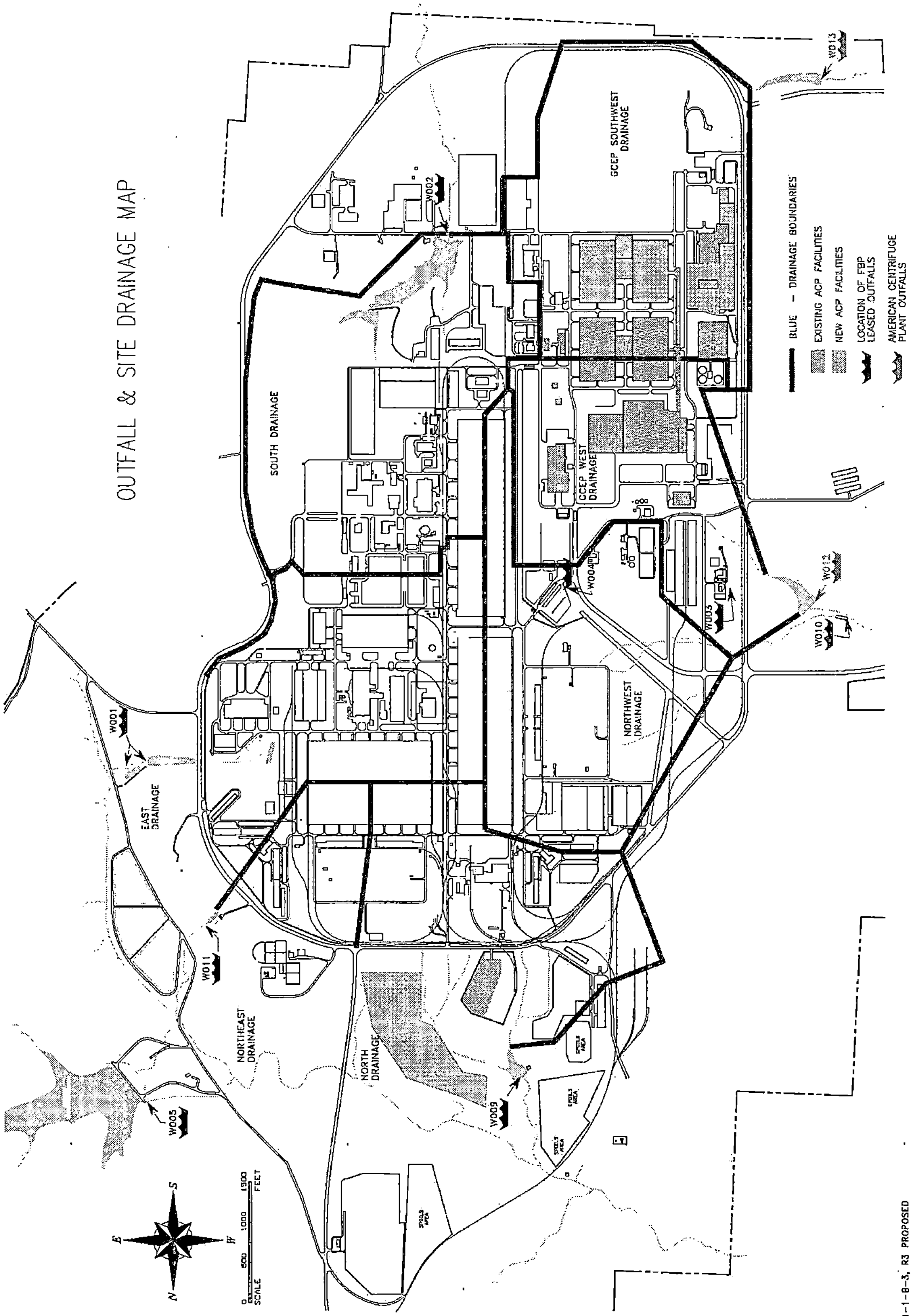

Figure 9.2-2 Locations of American Centrifuge Plant Outfalls Discharging to Waters of the United States

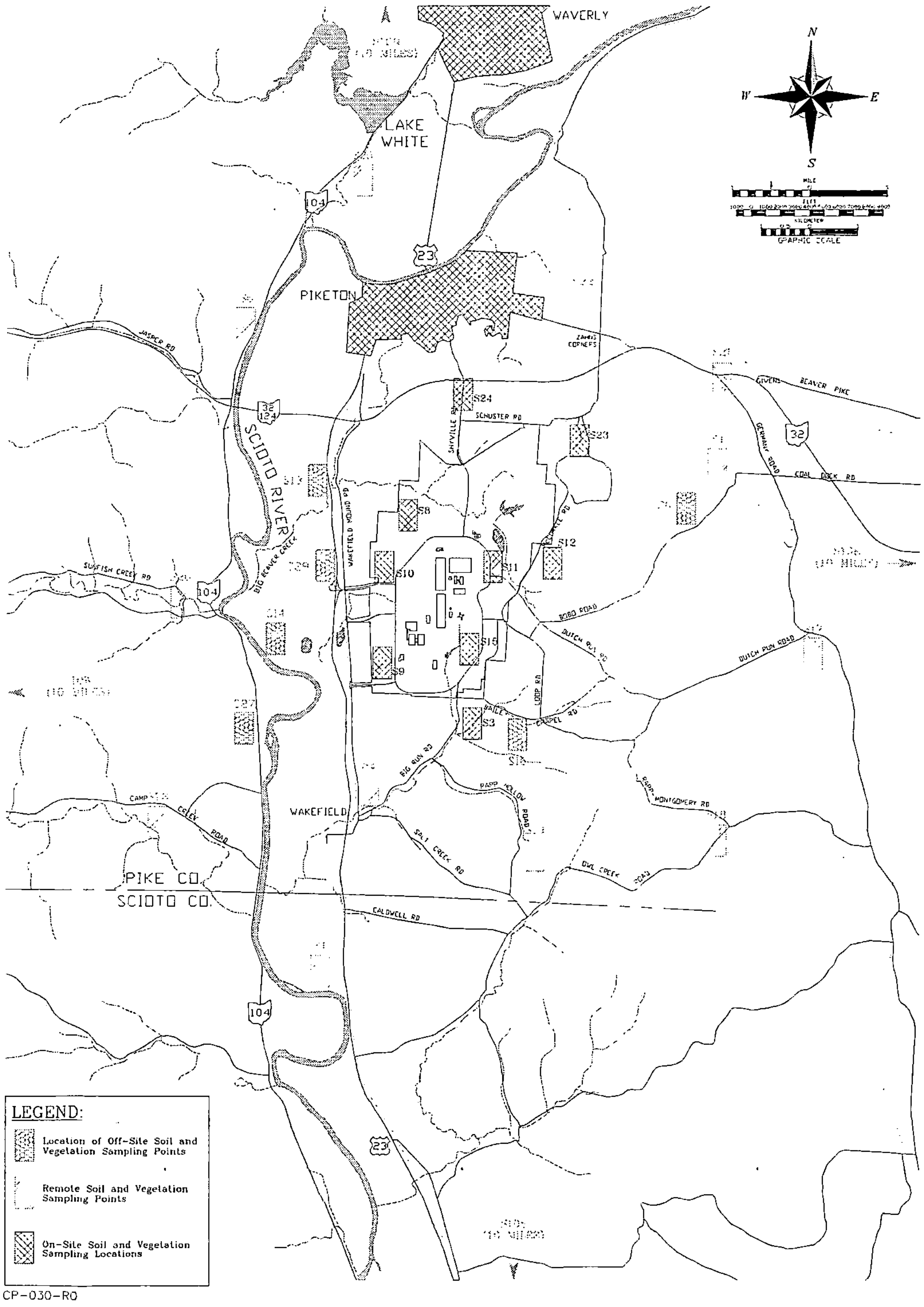

Figure 9.2-3 Locations of Soil and Vegetation Sampling Points

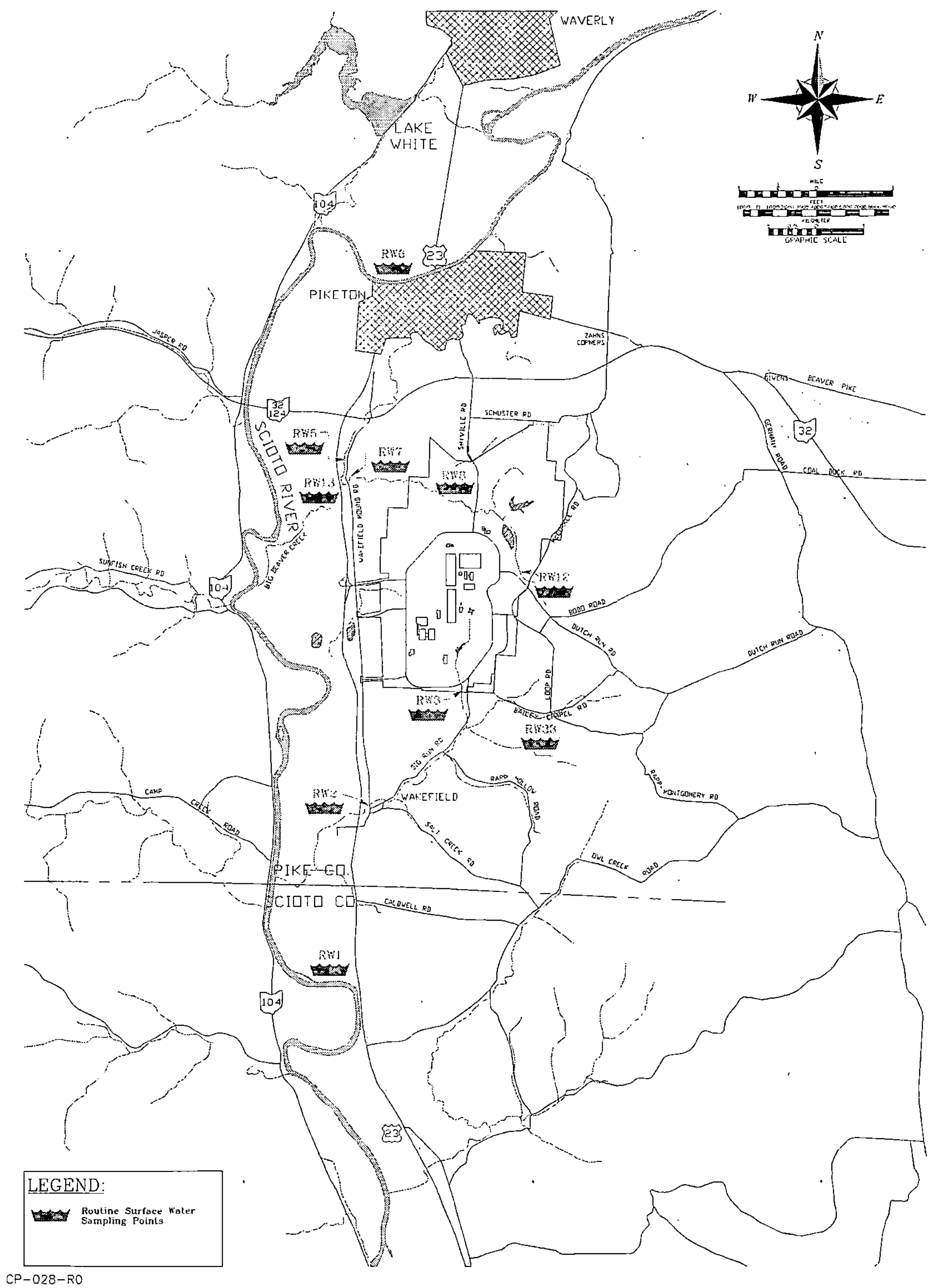

Figure 9.2-4 Locations of Surface Water Sampling Points

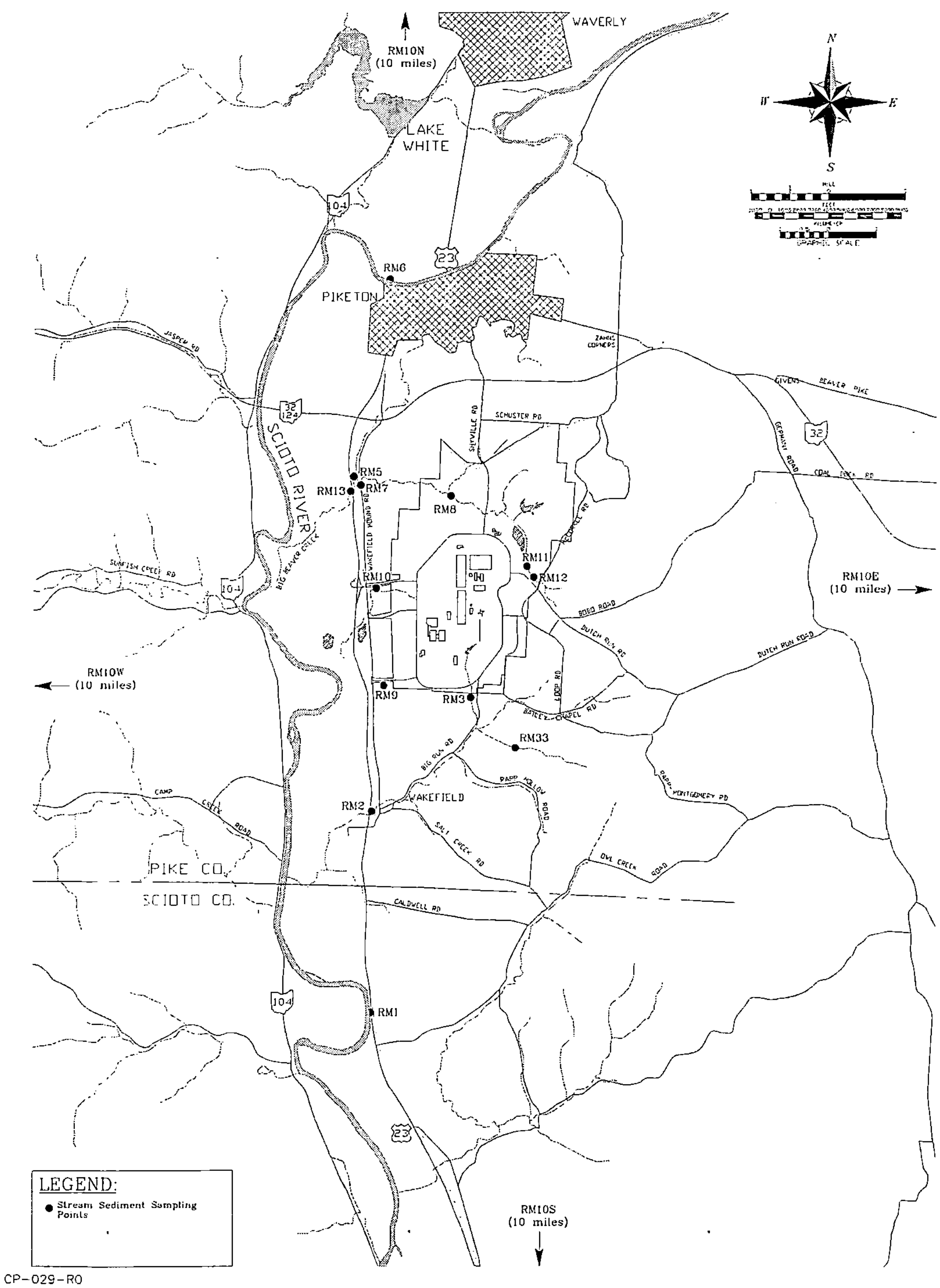

Figure 9.2-5 Locations of Stream Sediment Sampling Points

This figure is withheld pursuant to 10 CFR 2.390 and is located in Appendix B of this license application

Figure 9.2-6 Locations of Environmental Thermoluminescence Dosimeters on the U.S. Department of Energy Reservation

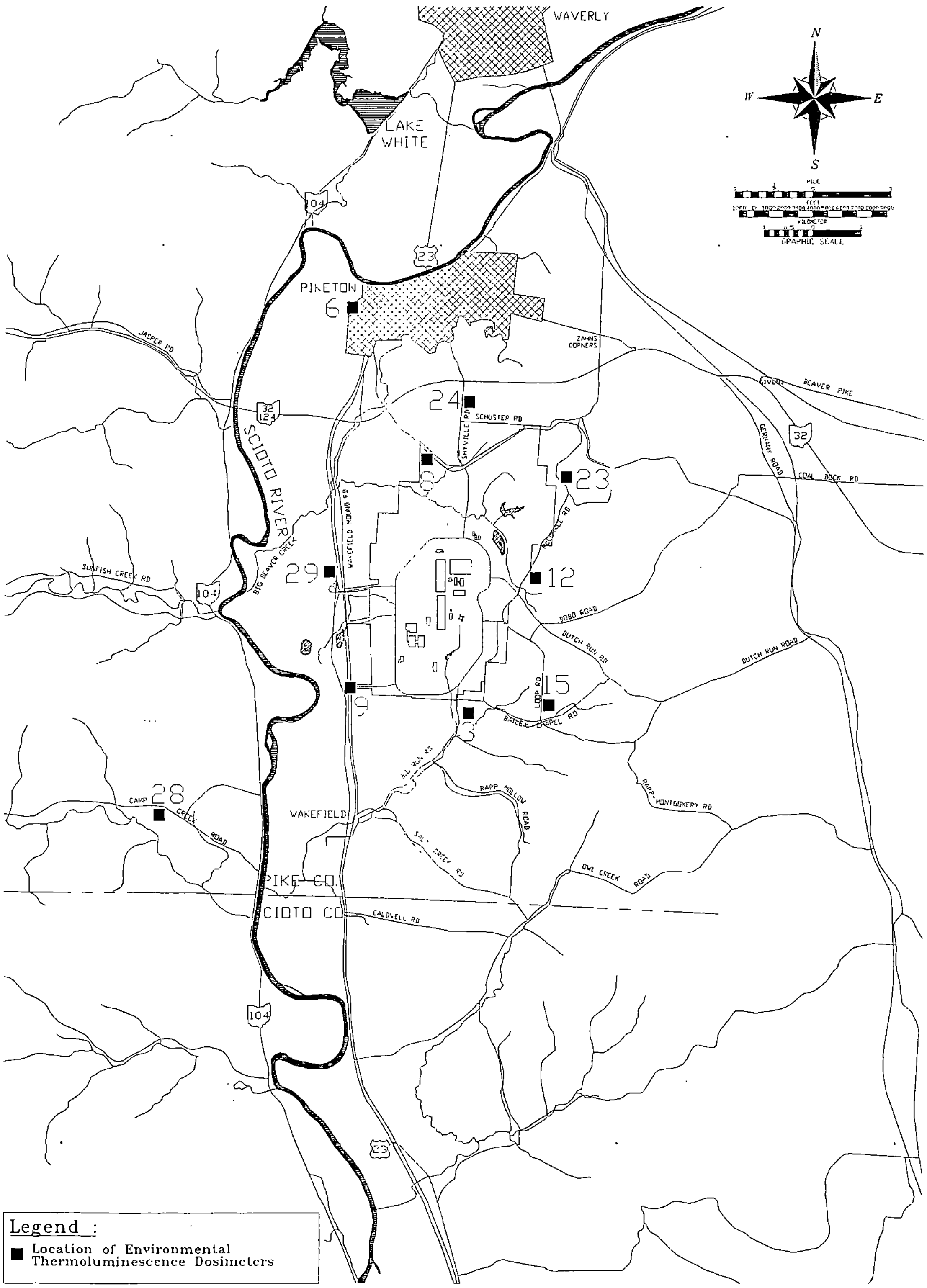

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Figure 9.2-7 Locations of Environmental Thermoluminescence Dosimeters Outside the U.S. Department of Energy Reservation Boundary

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10.0 DECOMMISSIONING

In accordance with NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications* (Revision 2), this chapter provides an overview of proposed decommissioning activities for the American Centrifuge Plant (ACP). The ACP is located in a leased area of the U.S. Department of Energy's (DOE) reservation in Piketon, Ohio.

10.1 High Assay Low Enriched Uranium (HALEU) Demonstration Program

The Licensee, American Centrifuge Operating, LLC (ACO or Corporation), is deploying a 16-machine AC100M HALEU cascade in leased areas under contract with the U.S. Department of Energy (DOE or Department). In support of this HALEU Demonstration Program, DOE amended the *Appendix 1 Lease Agreement between the U.S. Department of Energy and United States Enrichment Corporation for the Gas Centrifuge Enrichment Plant* (GCEP Lease Agreement). The amended GCEP Lease Agreement renewed and extended the term of the lease through May 31, 2022. Additionally, the amended GCEP Lease Agreement permits the construction and operation of the demonstration cascade by the Corporation (Licensee), the sublessee of the GCEP Lease and holder of the U.S. Nuclear Regulatory Commission (NRC) American Centrifuge Plant (ACP) Materials License.

The amended GCEP Lease Agreement includes the following statements pertaining to decommissioning liability:

- As of May 31, 2019, the Corporation (Licensee) had fully satisfied the lease turnover conditions and any existing financial assurance provided under Section 4.3 (of the GCEP Lease Agreement) was released, surety bonds were cancelled, and collateral returned to the Corporation (Licensee).

- Any facilities or equipment constructed or installed by the Corporation (Licensee) under the Demonstration Contract with the Department shall be included in Exhibit B (of the GCEP Lease Agreement) as Leased Personality and may be returned to the Department in an "as is" condition at the end of the lease term (May 31, 2022).

- The Department hereby assumes all liability for the decontamination and decommissioning of such facilities and equipment installed, and any work performed, under the Demonstration Contract with the Department including any materials or environmental hazards on the site. Therefore, no financial assurance for any liability or lease turnover conditions shall be required from the Corporation (Licensee).

- The parties agree that any work performed under the HALEU Demonstration Contract on the leased premises shall be considered a permitted use; any alternations or changes to the premises pursuant to the Demonstration Contract with the DOE shall be a permitted change to the premises; and any liabilities of the Corporation (Licensee) arising from or incident to the performance of work under the Demonstration Contract with the DOE shall be governed solely by such contract and any financial protection afforded to the Corporation (Licensee) as a person indemnified under the Act.

Pursuant to the modified DOE HALEU Contract, title to depleted uranium hexafluoride (UF_6) by-product (tails) from the HALEU enrichment process will be retained by DOE.

At the conclusion of the HALEU Demonstration Program, the facilities will be either returned to the Department in accordance with the requirements of the GCEP Lease Agreement or the Licensee will amend the ACP Materials License to allow phased implementation of expanded centrifuge enrichment cascades as described in Section 1.1.8 of the license application. At that time, a revised decommissioning funding plan, including an updated decommissioning cost estimate would be provided to the NRC for prior review and approval to reflect any new decommissioning liabilities.

10.2 American Centrifuge Plant (ACP) Decommissioning

The Licensee previously requested a 30-year license to operate the ACP. At the end of useful plant life, the ACP will be decommissioned such that the facilities will be either returned to the DOE in accordance with the requirements of the Lease Agreement with the DOE or will be released for unrestricted use. The criteria for final disposition of facilities will be established in the Decommissioning Plan (DP) which, as noted below, will be submitted prior to license termination. Nevertheless, for the purposes of the License Application for the ACP, the decommissioning discussions in this application and the decommissioning estimated costs are based on decontaminating the plant to the radiological criteria for unrestricted use in 10 *Code of Federal Regulations* (CFR) 20.1402. Information about the Licensee, the location of the site, and the types and authorized uses of licensed material are provided in Section 1.2 of the license application and a description of the site and immediate environs is provided in Section 1.3 of the license application.

Similar to the successful decommissioning efforts for the American Centrifuge Lead Cascade Facility, a more detailed DP for the ACP will be submitted by the Licensee in accordance with 10 CFR 30.36 (g), 10 CFR 40.42 (g), and 10 CFR 70.38(g) and applicable risk-informed NRC guidance provided in NUREG-1757, *Consolidated Decommissioning Guidance* (Volumes 1 - 3) prior to the time of license termination. Prior to decommissioning, an assessment of the radiological status of the ACP will be made. Enrichment equipment will be removed, leaving only the building shells and the plant infrastructure, including equipment that existed at the time of lease with the DOE (e.g., rigid mast crane, utilities, etc.). Classified material, components, and documents will be destroyed or disposed of in accordance with the *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant*. Requirements for nuclear material control and accountability will be maintained during decommissioning in a manner similar to the programs in force during ACP operation (NR-3605-0005). Depleted UF_6 material (tails), if not sold or disposed of prior to decommissioning, will be sold, or converted to a stable, non-volatile uranium compound and disposed of in accordance with regulatory requirements utilizing facilities constructed by DOE, as authorized by the *USEC Privatization Act*, and/or other licensed facilities. Radioactive wastes will be disposed of at licensed low-level waste disposal sites. Hazardous wastes will be treated or disposed of in licensed hazardous waste facilities.

The DP submitted at the time of license termination consists of several interrelated components, including (1) site characterization information, (2) remediation plan, and (3) a final status survey plan. The costs for activities required for these components have been identified in this chapter and estimated in the Decommissioning Funding Plan (DFP). Costs projected were developed based on the experience at the Portsmouth Gaseous Diffusion Plant during the transition to Cold Standby operation and decommissioning cost estimates developed for the American Centrifuge Demonstration Facility. Additionally, the Licensee had performed dismantling and decontamination work at the gaseous diffusion plants. Data and experience from these activities allowed a realistic estimation of expected decommissioning financial expenditures.

Using the cost data as a basis, financial arrangements are made to cover costs required to release the ACP for unrestricted use and to dispose of the tails. Updates on cost and funding will be provided periodically as described in Section 10.2.10.4. In accordance with 10 CFR 70.22(a)(9), 30.35, 40.36, and 70.25(a)(1), a DFP (NR-3605-0006) was previously submitted as part of the original license application for the ACP.

The following assumptions are utilized in the plan for decommissioning:

- No credit is taken for salvage value of equipment or materials.
- Decontamination liability is anticipated in the X-3001 and X-3002 Process Buildings, X-3012 Process Support Building, X-3344 Customer Services Building, X-3346 Feed and Withdrawal Building, X-3346A Feed and Product Shipping and Receiving Building, X-7725 Recycle/Assembly Building, X-7726 Centrifuge Training and Test Facility, X-7727H Interplant Transfer Corridor, X-2232C Interconnecting Process Piping, and miscellaneous cylinder storage yards.
- No decontamination is anticipated for the other ACP leased facilities.
- Decommissioning estimated costs are based on decontaminating the plant to the radiological criteria for unrestricted use in 10 CFR 20.1402.

The centrifuge assembly area in the X-7725 building is identified as the Decontamination Service Area (DSA). The centrifuge transport system is used to transport the centrifuges from the cascade area to the DSA.

The remaining sections of this chapter describe decommissioning plans and funding arrangements, and provide a detailed examination of the decontamination aspects of the program. The information herein was developed in connection with the decommissioning cost estimate and is provided for information. Specific elements of the planning may change with the submittal of the detailed DP required near the time of license termination.

The plan for decommissioning is to decontaminate or remove materials from the facilities promptly after cessation of ACP operations. Decommissioning planning begins by incorporating special design features into the plant. These features simplify dismantling and decontamination. The plans are implemented through proper management of Radiation Protection and Industrial

Health and Safety programs for the ACP. Decommissioning policies address radioactive waste management, physical security, and nuclear material control and accountability.

10.2.1 Decommissioning Design Features

Specific features are incorporated into the plant design to accommodate decontamination and decommissioning activities. The major features are described below.

10.2.1.1 Radioactive Contamination Control

The following features primarily serve to minimize the spread of radioactive contamination during operation, and simplify the eventual plant decommissioning. As a result, worker exposure to radiation and radioactive waste volumes are maintained as low as reasonably achievable (ALARA).

- Areas of the plant are sectioned off into clean areas and potentially contaminated areas, called Contamination Control Zones (CCZs) that have access control requirements. CCZs are buffer zones established where discrete areas of contamination might be occasionally encountered. Areas that are contaminated are called Contamination Areas (CAs). Figure 10.2.1-1 (located in Appendix B of this license application) provides a diagram showing the CCZ boundary. Procedures for these areas are encompassed by the Radiation Protection Program and serve to minimize the spread of contamination and simplify eventual decommissioning.

- Non-radioactive process equipment and systems are minimized in locations subject to likely contamination. This limits the size of the CCZs, and limits the activities occurring inside these areas.

10.2.1.2 Worker Exposure and Waste Volume Control

The following features primarily serve to minimize worker exposure to radiation and minimize radioactive waste volumes during decontamination activities. As a result, the spread of contamination is minimized as well.

- Ample access is provided for efficient equipment dismantling and removal of equipment that may be contaminated. This minimizes the time of worker exposure.

- Connections in the process systems are provided for thorough purging. This removes a significant portion of radioactive contamination prior to disassembly.

- Design drawings prepared for the plant, simplify the planning and implementing of decontamination procedures.

- Worker access to contaminated areas is controlled to assure that workers wear proper protective equipment and limit their time in the areas.

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

Figure 10.2.1-1 Commercial ACP Contamination Control Zone

10.2.2 Decommissioning Steps

Decommissioning may begin immediately following termination of operation, since only low radiation levels exist at this plant. Overall, the decommissioning is estimated to require approximately six years from plant shutdown to completion of the final status survey of radiological conditions. The order of activities to support decommissioning will generally be: planning and preparation; process system purging; equipment dismantling and removal; decontamination; disposition of equipment and material (including classified items); disposal of wastes; completion of a final status survey. The following sections provide an overview and explanation of each of these steps.

10.2.2.1 Overview

The intent of decommissioning is to return the ACP to an unrestricted use state. Removed equipment includes the centrifuges, the feed and withdrawal equipment, piping and components from systems providing UF_6 containment, systems in direct support of the centrifuges (e.g., cooling water), radioactive and hazardous waste handling systems, contaminated air filtration systems, etc. The remaining plant infrastructure includes utility services such as electrical power supply, sanitary water, fire suppression, ventilation, communications, and sewage treatment.

Decontamination of the plant will not require the installation of a new facility dedicated for that purpose since the X-7725 building will serve as the DSA and will accommodate repetitive equipment decontamination of centrifuges and other components. The DSA is described in Section 10.2.8.1 of this license application and will be the location for decontamination activities.

Although certain unclassified components may be reused or sold as scrap, for conservatism this plan assumes only that components will be decontaminated in accordance with radiation protection requirements. Classified parts will be dispositioned in accordance with an approved Security Plan. Table 10.2.2-1 of this license application lists components for potential decontamination at decommissioning.

The Licensee intends to evaluate possible commercial uses of UF_6 tails. UF_6 tails which are not commercially reused will be converted to a stable form and disposed of in accordance with the *USEC Privatization Act* and other applicable statutory authorizations and requirements at DOE's UF_6 conversion facilities and/or other licensed facilities. UF_6 tails are stored in steel cylinders until the tails material can be processed in accordance with the disposal strategy established by the Licensee. The Licensee provides financial assurance to fund the estimated cost of conversion and disposal of the depleted uranium inventory as it is generated during operation. This funding is described in the DFP and is in addition to the funding requirements for decommissioning the ACP. At full capacity, the ACP will generate approximately 8,400 Metric Ton (MT) of UF_6 tails annually. Over the 30-year license, that is a total of approximately 214,400 MT of UF_6 tails, as noted in Table C3.19 of the DFP. Depending on technological developments and the existence of facilities available prior to ACP shutdown, the tails may have commercial value and may be marketable for further enrichment or other processes. However, funding provisions are made to dispose of the tails should that become necessary.

Contaminated portions of the buildings will be decontaminated. Structural contamination is expected to be limited to the areas indicated on Figure 10.2.1-1 (located in Appendix B) inside the CCZ of the plant. The remainder of the ACP is not expected to require decontamination. Good housekeeping practices during normal operation and cleanup activities following spills or contamination events will maintain these other areas contamination free. Decontamination activities will continue until facilities satisfy the specified radiological criteria.

10.2.2.2 Purging

At the end of useful operation, the ACP is shut down and UF_6 material is removed to the fullest extent possible by normal process operation. This is followed by evacuation and purging of process systems. This shutdown and purging portion of the decommissioning process is estimated to take approximately three months.

10.2.2.3 Dismantling and Removal

Dismantling is the process of unbolting, disconnecting, cutting, etc., of components requiring removal. The dismantling and removal activities are simple but labor intensive. They generally require the use of protective equipment. The work process will be optimized, considering the following:

- Minimize spread of contamination and the need for protective equipment;
- Balance the number of cutting and removal operations with the resultant decontamination and disposal requirements;
- Optimize the rate of dismantling with the rate of decontamination plant throughput;
- Provide storage and laydown space required, as impacted by retrievability, criticality safety, security, etc.; and
- Balance the cost of decontamination with the cost of disposal.

Details of the complex optimization process will be decided near the end of plant useful life, taking into account specific contamination levels, market conditions, and available waste disposal sites. To avoid laydown space and contamination problems, dismantling will proceed generally no faster than the downstream decontamination process. The time frame to accomplish both dismantling and decontamination is estimated to be five years.

10.2.2.4 Decontamination

The decontamination process is addressed separately in Section 10.2.8 of this chapter. The decommissioning estimated costs are based on decontaminating the plant to the radiological criteria for unrestricted use in 10 CFR 20.1402.

10.2.2.5 Salvage and Sale

Items to be removed from the facilities can be categorized as potentially re-usable equipment (whether contaminated or decontaminated), recoverable decontaminated scrap, and wastes. Based on a 30-year plant operating life, operating equipment is not assumed to have a significant reuse value. Uranium-bearing equipment that remains in the plant will be treated and disposed of appropriately. Smaller amounts of steel, copper, and other metals can be recovered and sold at market price. However, for conservatism, no credit is taken for salvage value in the DFP.

Other items are considered waste. Wastes have no salvage value.

10.2.2.6 Disposal

Wastes produced during decommissioning will be collected, handled, and disposed of in a manner similar to that described for those wastes produced during normal operation. Wastes will consist of normal industrial trash, non-hazardous chemicals and fluids, small amounts of hazardous materials, and low-level mixed (LLMW) and radioactive (LLRW) wastes. The radioactive waste will primarily be crushed centrifuge rotors, trash, and citric cake. Citric cake consists of uranium and metallic compounds precipitated from citric acid decontamination solutions. It is estimated that approximately 76,388 cubic feet of compacted radioactive waste will be generated during the decommissioning operation. This waste may be subject to further volume reduction prior to disposal.

Radioactive wastes (both LLRW and LLMW) will ultimately be disposed of in licensed low-level radioactive waste disposal facilities (this includes radioactive source and byproduct material sources). Hazardous wastes will be disposed of in hazardous waste disposal facilities. Non-hazardous and non-radioactive wastes will be disposed of in a manner consistent with good industrial practice and in accordance with applicable regulations. A more complete estimate of the wastes and effluent to be produced during decommissioning will be provided in the DP to be submitted at or about the time of license termination.

The ultimate disposal of UF_6 tails remains to be determined between potential commercial uses or processing at the DOE UF_6 conversion facility in Piketon, Ohio. However, for conservatism, the Licensee provides financial assurance to fund the estimated cost of conversion and disposal of the depleted uranium inventory. This funding is described in the DFP and is in addition to the funding requirements for decommissioning the ACP. Classified components and documents will be disposed of in accordance with the requirements of the *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant.*

10.2.2.7 Final Status Survey

A final status survey of the radiological conditions of the plant is performed to verify proper decontamination. The evaluation of the final radiation survey is based, in part, on an initial radiation survey performed prior to operation. The initial survey determines the background radiation of the area; providing a datum for measurements that determine any increase in levels of radioactivity.

The final status survey will systematically take measurements and perform sampling to describe radioactivity over the ACP. The intensity of the survey will vary depending on the location (i.e., the buildings, the immediate area around the buildings, the controlled fenced area, and the remainder of the site). The survey procedures and results will be documented in a report. The results of the report will become part of the application to terminate the license. The format and content of the report will follow current NRC guidance (Section 4.5 of Volume 2 of NUREG-1757).

Table 10.2.2-1 Components for Potential Decontamination/Disposal at Decommissioning

Components	Description [units]	Estimated Quantity
Centrifuges [1, 2]	Internals: Rotor Assemblies, Motors, Suspensions and Mounts (Classified)	12,000
Service Modules [2]	Structural Components	0
Piping	Less than 1 in. Process Piping length (Lft) Includes Tubing [3]	0
	1-16 in. Process Piping length (Lft)	271,840
Blowers	Feed/Withdrawal Exhaust Blowers	2
Pumps	Vacuum (Evacuation/Purge); RHW Pumps	119
Ventilation	Ductwork; Miscellaneous WISP Ducting (ft^3) [3]	3,677
Surfaces	Building Floors, Yards, Equipment (ft^2) [4]	2,494,819
Valves	Process valves and MIVs (excluding Sheetmetal)	18,631
	Miscellaneous valves	1,385
Sources	Source and byproduct material sources used at the Lead Cascade	11
Process Equipment	Feed Ovens, Autoclaves, Cold Boxes	91
Cranes	Ridge Mast (RMC), Bridge, Wall and Jib Cranes; Cylinder Transporters, Trolleys	29
Scales	Process Weighing Equipment	12
Compressors	Process Gas Compressors	4
Heat Exchangers (HX)	Machine Cooling Water HX, Freezer/Sublimers, Tails Coolers	36
Traps	Chemical traps (8 banks of 4), Cold Traps, Roughing Filters, Miscellaneous Traps	71
Tanks (UF_6)	Holdup, Surge, and Dump Tanks	3
Upender	Trailer Upender (X-7725)(ft^3)	3
Cylinders	Tails – 48G/48X (14, 10 Ton)	17,191
Cylinders	Product and Feed (2.5 Ton) Gas Test Area (12B)	450
Other Equipment	UF_6 Portable Carts; Buffer Storage Stands; Mass Spectrometers; Contaminant Monitors; Miscellaneous Platforms; and Gas Test Stand Center (GTC) Stand Structures	69

Note 1: Amount includes 11,520 operational units plus 480 contaminated spare centrifuges.
Note 2: Centrifuge casings and service module structural steel is not considered waste. These items are to be removed, disassembled, decontaminated to NRC 'Free Release' criteria, and stored for later disposition.
Note 3: Piping <1" (assumed to be instrument piping/tubing), ventilation ductwork, and heat exchangers are assumed to not be internally contaminated. Therefore, these components can be externally decontaminated and remain as part of the building Balance of Plant.
Note 4: Amount of wall area (ft^2) not provided, because it is not anticipated to need decontamination at the time of decommissioning.
Note 5: Equipment re-utilized from operational phase (not new or purchased).
Note 6: Equipment procured (see Table C3.15 of the Decommissioning Funding Plan for the ACP).

Components	Description [units]	Estimated Quantity
Decontamination Equipment	Centrifuge Transporter[5]	2
	Cranes (Process Area - RMC)[5]	8
	Cranes, Bridge X-7725[5]	2
	Centrifuge Mobile Equipment[5]	4
	Centrifuge Dismantling Equipment (6/X-7725 and 2/X-7726 Assembly Stands)[6]	8
	Cutting Machines[6]	2
	Degreasers[6]	2
	Decontamination Tanks[6]	4
	Wet Blast Cabinets[6]	2
	Crusher[6]	1

10.2.3 Management/Organization

Management of the decommissioning program will assure proper training and procedures are provided to assure worker health and safety. The programs will focus on minimizing waste volumes and worker exposure to hazardous or radioactive materials. Qualified contractors assisting with decommissioning will be subject to ACP security and training requirements, and procedural controls.

10.2.4 Health and Safety

Consistent with the policy during ACP operation, the policy during decommissioning is to keep individual and collective occupational radiation exposure with the ALARA principle. A Radiation Protection Program will identify and control sources of radiation, establish worker protection requirements and direct the use of survey and monitoring instruments.

10.2.5 Waste Management

Radioactive and hazardous wastes produced during decommissioning will be collected, handled, and disposed of in accordance with regulations applicable to the ACP at the time of decommissioning. Generally, procedures will be similar to those described for wastes produced during operation. These wastes will ultimately be disposed of in licensed radioactive or hazardous waste disposal facilities. Non-hazardous and non-radioactive wastes will be disposed of consistent with good industrial practice, and in accordance with applicable regulations.

10.2.6 Security and Nuclear Material Control

Requirements for physical security and for nuclear material control and accountability will be maintained during decommissioning in a manner similar to the programs in force during ACP operation. This includes requirements for control of classified information and classified equipment described in the *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant* and the requirements for control of nuclear materials in the *Fundamental Nuclear*

Material Control Plan for the American Centrifuge Plant. The DP is submitted near the end of plant life and will provide a description of revisions to these programs.

10.2.7 Record Keeping

Records important for safe and effective decommissioning of the ACP are maintained in accordance with established Records Management and Document Control procedural requirements. Information maintained in these records include:

- Records of spills or other unusual occurrences involving the spread of contamination in and around the plant, equipment, or site. Records of spills or other unusual occurrences may be limited only to instances when contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as concrete. These records will include any known information on identification of involved radionuclides, quantities, forms, and concentrations;

- As-built drawings and modifications of structures and equipment in areas where radioactive materials are used and/or stored, including locations that possibly could be inaccessible (e.g., buried pipes which may be subject to contamination); and

- A list contained in a single document that is updated every two years of the following:

  - Areas designated and formerly designated as restricted areas as defined under 10 CFR 20.1003.

  - Areas outside of restricted areas that require documentation under 10 CFR 70.25(g)(1).

  - Areas outside of restricted areas where current and previous wastes have been buried as documented under 10 CFR 20.2108.

  - Areas outside of restricted areas that contain material such that, if the license expired, the Licensee would be required to either decontaminate the area to meet the criteria for decommissioning in 10 CFR Part 20, Subpart E or would apply for NRC approval for disposal under 10 CFR 20.2002.

- Records of the cost estimate performed for the DFP, and records of the funding method used for assuring funds, including a copy of the financial assurance mechanism and any supporting documentation.

10.2.8 Decontamination

The DSA, the general procedures used to decontaminate, and the expected results of decontamination are described in the paragraphs below. Table 10.2.2-1 lists the major components and structures that may need to be decontaminated to some extent at the plant. Other components and structure will generally not require any decontamination. The Licensee anticipates low amounts and areas of actual contamination due to strict adherence to ALARA principles throughout the plant's life.

There are two general methods of decontamination, which may be used to decontaminate the ACP: dry and wet. Dry involves using an always safe vacuum cleaner (vacuuming), scooping up the material with a dust pan (low abrasive materials), sweeping material up with a brush or broom, or high abrasive (chipping or wire brush). Wet decontamination involves using films of cleaning solutions with mops, squeegees, rags, or dip tanks. Although wet decontamination or a dry decontamination variation, such as dry ice blasting, may be utilized for decontamination of the ACP, these methods are not anticipated to be utilized to a significant extent, and, therefore, are not included in the DFP estimate. For decontamination and decommissioning of the ACP and establishing the associated funding, it is assumed that a dry decontamination process is utilized throughout. The actual decontamination method or methods to be utilized to decontaminate and decommission the ACP will be established based upon the site characterization survey performed during the decommissioning planning and preparation phase and will be described in the Decommissioning Plan.

The DFP estimate does consider scarifying, to a 1/8-inch depth, the cylinder yard areas in their entirety as a conservative action. Any time surfaces are disturbed, such as with scarifying concrete, there is a potential to produce airborne radioactivity. To mitigate these concerns, airborne monitoring for the personnel performing the work would be provided, these individuals would be included in the internal monitoring program (urinalysis), and if the conditions exist, respiratory protection may be required. Furthermore, scarifying equipment may use a water spray to minimize dust, cool the cutting wheels, or use a limited amount of water as a media, but this is not considered to be a liquid waste as it is anticipated to evaporate to leave a dry debris for solid waste disposal.

10.2.8.1 Decontamination Service Area

The centrifuge assembly area within X-7725 building is identified as the DSA. The centrifuge transport system would be used to transport the centrifuges from the process buildings to the DSA. The DSA handles centrifuges, feed, withdrawal, sampling and transfer equipment to be disassembled and dispositioned along with the UF_6 vacuum pumps, valves, piping, and other miscellaneous equipment. Unusable material will be destroyed. The DSA will have four functional areas: disassembly area, buffer stock area, decontamination area, and scrap storage area. Equipment in the decontamination area may include:

- Transport and manipulation equipment
- Dismantling area
- Cutting machines
- Dismantling boxes and tanks (e.g., B-25 boxes)

- Degreasers
- Citric acid and demineralized water baths
- Contamination monitors
- Wet blast cabinets
- Crushers or size reduction equipment
- Shredding equipment
- Scrubbing facility

There is no normal operational need for the ACP to have a decontamination facility readily available.

10.2.8.2 Procedures

Procedures for decontamination will be developed and approved by plant management to minimize worker exposure and waste volumes, and to assure work is carried out in a safe manner. At the end of useful plant life, some of the equipment, most of the buildings, and the outdoor areas should already be acceptable for release for unrestricted use in accordance with 10 CFR 20.1402. If these areas were inadvertently contaminated during ACP operation, they would likely be cleaned up when the contamination is discovered. This limits the scope of necessary decontamination at the time of decommissioning.

The centrifuges will be processed and the following operations will be performed:

- Removal of external fittings;
- Removal of bottom flange, motor and bearings, and collection of contaminated oil;
- Removal of top flange, and withdrawal and disassembly of internals;
- Degreasing of items, as required; and
- Destruction of classified parts by shredding, crushing, burial, etc.

10.2.8.3 Results

Recoverable items will be externally decontaminated and suitable for reuse except for a very small amount of internally contaminated items where recovery and reuse is not feasible. There is potentially a small amount of salvageable scrap material. Material requiring disposal will be process piping, trash, and residue from the effluent treatment systems. No problems are anticipated which will prevent the facilities from being released for unrestricted use.

10.2.9 Agreements with Outside Organizations

The decommissioning activities described herein and in the DFP provide for decontamination of the ACP for unrestricted use. As such, no agreements with outside organizations are required for control of access to the plant following shutdown and decommissioning.

10.2.10 Arrangements for Funding

This section provides a general estimate of plant decommissioning costs and UF_6 tails disposition costs, as well as explains the arrangements made to assure funding is available to cover these costs. A more detailed description of these costs and the financial assurance mechanism is provided in the DFP.

10.2.10.1 Plant Decommissioning Costs

Table 10.2.10-1, provides a summary of the cost estimates of the major decommissioning activities described in Section 10.2.2. Costs are provided in 2008 dollars with a 25 percent contingency factor added based on the NRC guidance (Volume 3 of NUREG-1757). As noted below, the total estimated cost to decommission the 3.8 million SWU ACP, excluding UF_6 tails disposition, is $377.3 million. Since costs will likely change between the time of license issuance and actual decommissioning, the Licensee will adjust the cost estimate annually prior to operation of the facility at full capacity, and after full capacity is reached, no less frequently than every three years consistent with the requirements of 10 CFR 70.25(e) and recent NRC changes to financial assurance requirements for materials licensees (Federal Register, Volume 192). The method for adjusting the cost estimate will consider the following:

- Changes in general inflation (e.g., labor rates, consumer price index);
- Changes in price of goods (e.g., packing materials);
- Changes in price of services (e.g., shipping and disposal costs);
- Changes in plant condition or operations; and
- Changes in decommissioning procedures or regulations.

These costs are estimated as explained below:

Planning and Preparation: $3.3 million

Scope to be completed in one year and includes developing and submitting a detailed DP as a license amendment for NRC review and approval. Activities anticipated during this phase include:

- Develop Project Execution Plan and Schedule (including the organization and staffing plan and needed services);

- Develop and submit the Decommissioning Plan;
- Develop/implement Site Characterization Plan;
- Review/approve Site Decommissioning Plan by the NRC;
- Develop Decommissioning Activity Procedures; and
- Design Decommissioning Service Area (DSA).

Decontamination and/or Dismantling of Radioactive Facilities: $51.5 million

This is based upon utilizing salary and hourly workers at their respective average cost over a five-year duration. For conservatism, decommissioning estimated costs are based on decontaminating the plant to the radiological criteria for unrestricted use in 10 CFR 20.1402. Activities anticipated during this phase include:

- Prepare the decontamination Service Area;
- Internal decontamination of facilities;
- Dismantle centrifuges to include waste segregation and staging;
- Dismantle facilities and components; and
- Tails cylinder movement/disposition to include material title transfer to DOE.

Restoration of Contaminated Areas On Plant Grounds: $0.9 million

This is based upon utilizing salary and hourly workers at their respective current average cost distribution over a two-year duration. This assumes the contamination of the plant grounds from the ACP operations will be minimal. Activities anticipated during this phase include:

- External decontamination of facilities;
- Perform Health Physics surveys;
- Scarify cylinder storage yard surfaces; and
- Collect/dispose of yard debris.

Final Status Survey: $1.6 million

This is based upon utilizing salary technicians at their current average cost distribution for a period of 2.5 years. Costs do not include any NRC confirmatory surveys to verify the results of the Final

Status Survey. Activities anticipated during this phase include:

- Develop/implement survey plans;
- Collect/analyze data;
- Perform confirmatory surveys;
- Develop final survey report; and
- Prepare License Amendment to terminate the license.

Site Stabilization and Long-Term Surveillance: $3.0 million

As previously stated, the intent of decommissioning is to return the plant to the radiological criteria for unrestricted use. To accomplish this activity, stabilization and surveillance is required due to the number of components involved and the duration of the decommissioning effort. This scope of work occurs throughout the six year decommissioning period and involves maintenance and surveillance activities on IROFS, as required, until the license is terminated.

Packing Materials, Shipping, and Waste Disposal: $61.6 million

This is based upon shipping and disposal of the internals for 12,000 centrifuges (which includes operating centrifuges as well as contaminated spares), feed and withdrawal equipment, and other components totaling approximately 76,388 cubic feet of solid waste, 16,225 gallons of liquid waste from the centrifuge internals and 1,728,000 cubic feet of classified waste in non-reusable packaging.

Equipment and Supply: $19.6 million

This includes the purchase or lease of dismantling, cutting, degreasing, and crushing equipment; decontamination tanks, wet blast cabinets, and over 20,000 containers (i.e., B-25 boxes and 55 gallon drums).

Laboratory: $1.5 million

This includes labor costs for sampling, transport, testing, and analysis of samples.

Indirect Services: $71.9 million

This includes support services (such as laundry, janitorial, etc.) and infrastructure costs (such as water, power, etc.) not included in other tasks.

Miscellaneous: $41.6 million

This includes direct costs of $2.9 million for miscellaneous material for decommissioning and $38.7 million for indirect costs, such as NRC review fees for the submitted DP, license fees, DOE lease fees, and business insurance.

Subtotal	**$256.5 million**
General and Administrative (6 percent)	**$15.4 million**
Contractor Profit (15 percent)[1]	**$29.9 million**
Contingency (25 percent)	**$75.5 million**
Total Plant Decommissioning Cost Estimate	**$377.3 million**

[1] Contractor Profit = 0.15[Subtotal + General and Administrative - Other Indirect Costs (excluding insurance) – Outside Services portion of the Packaging, Shipping, and Waste Disposal Costs]

10.2.10.2 UF_6 Tails Disposition Costs

Cost estimates to dispose of UF_6 tails generated during ACP operation are separate from the cost estimates to decommission the plant. As noted previously, the ultimate disposal of UF_6 tails remains to be determined. The Licensee intends to evaluate possible commercial uses of UF_6 tails before having the tails processed by the DOE UF_6 conversion facility in Piketon, Ohio. UF_6 tails are stored in steel cylinders until they can be processed in accordance with the disposal strategy established by the Licensee. Depending on technological developments and the existence of facilities available prior to ACP shutdown, the tails may have commercial value and may be marketable for further enrichment or other processes. However, for the purposes of calculating the UF_6 tails disposition cost, the Licensee assumes that the total quantity of tails generated during ACP operation are processed by the DOE UF_6 conversion facility in Piketon, Ohio.

For conservatism, the Licensee provides financial assurance to fund the estimated cost of conversion and disposal of the depleted uranium inventory as it is generated during ACP operation. This funding is described in the DFP and is in addition to the funding requirements for decommissioning the ACP. As with plant decommissioning, the cost estimate will likely change between the time of license issuance and actual decommissioning. The Licensee commits to adjust the cost estimate for tails disposal annually. The method for adjusting the cost estimate will consider the same factors as previously described in Section 10.2.10.1 of this chapter.

At full capacity, the ACP will generate approximately 8,400 MT of UF_6 tails annually. As with other decommissioning costs, the disposal cost estimate for UF_6 tails disposal is provided in 2008 dollars. Consistent with the recommendation in the NRC's guidance on decommissioning (Section A.3.1.2.3 of Volume 3 of NUREG-1757), a 25 percent contingency factor is applied to the tails disposal cost estimate. The total estimated cost to dispose of UF_6 tails over the 30-year license, including a four-year ramp up to full capacity and the 25 percent contingency factor, is $896.9 million. The basis for this estimate is provided in the DFP.

10.2.10.3 Total Decommissioning Liability

The Licensee's total decommissioning liability is the sum of the total plant decommissioning costs and the tails disposition costs. The Licensee's total liability for decommissioning the ACP, including applicable contingencies, is:

Plant Decommissioning Cost	\$ 377.3 million
UF_6 Tails Disposition Cost	\$ 896.9 million
Total Decommissioning Liability	\$1,274.2 million

10.2.10.4 Funding Arrangements

Per the exemption request in Section 1.2.5 of this license application, the financial assurance for a portion of the decommissioning costs to include disposition of centrifuges and UF_6 tails will be provided incrementally as centrifuges are built/installed and UF_6 tails generated. The modular aspect of the American Centrifuge technology allows enrichment operations to begin well before the full capacity of the plant is reached. Thus, the decommissioning liability for centrifuges and UF_6 tails is incurred incrementally as more centrifuges, and associated equipment, are added to the process, until such time as full capacity of the facility (i.e., 3.8 million SWU) is achieved. Once full capacity of the facility is achieved, the UF_6 tails are generated at a relatively constant rate throughout the life of the plant.

Full funding for decommissioning of the facilities will be provided in the initial executed financial assurance instrument. To ensure adequate financial assurance is in place as centrifuges, and associated equipment, are added to the process and placed into operation, the Licensee will forecast and update the cost estimates and provide a revised funding instrument to NRC annually to cover the upcoming year of operation. This incremental funding approach will be utilized until operation at full capacity. Once full capacity of the facility is achieved, the Licensee will annually adjust the cost estimate for UF_6 tails disposal and all other decommissioning costs will be adjusted periodically, and no less frequently than every three years. In this way, financial assurance will be made available as the decommissioning liability is incurred. This exemption is justified based on the unique modularity aspects of centrifuge technology that allow enrichment operations to begin well before the full capacity of the plant is reached. In addition, the NRC has accepted an incremental approach to funding disposal cost of tails for the gaseous diffusion plants. Financial assurance will be provided in the form of a surety method or other guarantee method as required by 10 CFR 70.25(f). The selected guarantee method is described in the DFP, included as part of this license application. In the DFP, methods are described for periodic adjustments in the cost estimate and resulting necessary adjustments to the funding method.

10.3 References

1. *Appendix 1 Lease Agreement between the U.S. Department of Energy and United States Enrichment Corporation for the Gas Centrifuge Enrichment Plant* (GCEP Lease Agreement), Amendment dated May 31, 2019

2. Federal Register, Volume 68 Number 192, *Financial Assurance for Materials Licensees*, Final Rule, October 3, 2003

3. HALEU Demonstration Contract Number 89303519CNE000005, awarded May 31, 2019 and definitized on October 31, 2019

4. NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications,* Revision 2

5. NUREG-1757, Consolidated Decommissioning Guidance, Volume 1, , *Decommissioning Process for Materials Licensees,* Revision 2

6. NUREG-1757, Consolidated Decommissioning Guidance, Volume 2, *Characterization, Survey, and Determination of Radiological Criteria,* Revision 1

7. NUREG-1757, Consolidated Decommissioning Guidance, Volume 3, *Financial Assurance, Recordkeeping, and Timeliness,* Revision 1

8. NR-3605-0005, *Fundamental Nuclear Material Control Plan for the American Centrifuge Plant*

9. NR-3605-0006, *Decommissioning Funding Plan for the American Centrifuge Plant*

10. SP-3605-0041, *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant*

11. W. Brown (DOE) letter to Mr. Phil Sewell (USEC), *Conversion and Disposal of Depleted Uranium Hexafluoride (DUF6) Generated by USEC at the American Centrifuge Plant in Piketon, Ohio*, dated February 10, 2006

Table 10.2.10-1 Plant Decommissioning Cost Estimates and Expected Duration

Task/Item	Cost Estimate (Millions, 2008 dollars)	Approx. Percentage
Planning and Preparation	$3.3	1%
Decontamination and/or Dismantling of Radioactive Facilities	$51.5	20%
Restoration of Contaminated Areas On Plant Grounds	$0.9	1%
Final Status Survey	$1.6	1%
Site Stabilization and Long-Term Surveillance	$3.0	1%
Packing Materials, Shipping, and Waste Disposal	$61.6	24%
Equipment and Supply	$19.6	8%
Laboratory	$1.5	1%
Indirect Services	$71.9	28%
Miscellaneous	$41.6	16%
Subtotal	$256.5	100%
General and Administrative (6%)	15.4	
Contractor Profit (15%)	29.9	
Contingency (25%)	$75.5	
Total Plant Decommissioning Cost	**$377.3**	
UF_6 Tails Disposal Costs **UF_6 Tails Contingency (25%)**	**$717.6** **$179.4**	
Total UF_6 Tails Disposition Cost	**$896.9**	
Total Decommissioning Liability	**$1,274.2**	

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11.0 MANAGEMENT MEASURES

Management measures are functions that are applied to items relied on for safety (IROFS) to provide reasonable assurance that the IROFS are available and reliable to perform their functions when needed. 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. Management measures are applied in a graded approach. The degree to which management 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 Licensee will periodically review IROFS per the requirements of 10 CFR 70.62(a)(3) to ensure their availability, reliability, and have not changed. As the final design is developed for the American Centrifuge Plant (ACP), the management system and design approach will require that the final designs be reviewed against the ISA to ensure the ISA is bounding.

As discussed in Section 1.1.8 of this license application, American Centrifuge Operating, LLC's (ACO) long-term goal is to resume commercial enrichment production consistent with market demand. The ACP design is modular, with the basic building block of enrichment capacity being a cascade of centrifuges. Modular deployment would accommodate market demand on a scalable, economical gradation. As such, the Management Measures will be implemented to support the modular deployment.

The next phase of enrichment production includes the deployment of a cascade of 16 centrifuges to demonstrate production of high-assay, low-enriched uranium (HALEU) fuel for advanced reactors. The primary building/facilities directly involved in HALEU Demonstration are the X-3001 Process Building, X-3012 Process Support Building, X-7725 Recycle/Assembly Building, X-7726 Centrifuge Training and Test Facility, and X-7727H Interplant Transfer Corridor. The Licensee will notify NRC well in advance of the transition into any future phases of ACP deployment. For further plant and process specifics related to the HALEU Demonstration Program, refer to LA-3605-0003A, *Addendum 1 of the ISA for the American Centrifuge Plant – HALEU Demonstration.*

The general use of the term ACP in the remainder of this chapter is intended to refer to both the commercial ACP operation and the HALEU Demonstration. HALEU Demonstration will be specifically noted, as necessary, when the context is uniquely applicable to HALEU Demonstration.

11.1 Configuration Management

The Configuration Management (CM) Program for the American Centrifuge Plant (ACP) 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 plant baseline configuration are identified and controlled to help ensure safety through consistency among the plant design and operational requirements, the physical configuration, and the plant documentation. The 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 plant's configuration.

The goal of the CM program is to ensure that the ACP has accurate, current documentation that matches the plant's physical/functional/operational configuration, while complying with applicable requirements.

11.1.1.1 Program Overview

The Piketon Engineering Manager has primary responsibility for the implementation of the CM Program for the ACP. The CM Program is applicable to the plant, 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 plant, 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) through (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, Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration, 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 ACP manager and functional area:

11.1.1.2.1 Piketon Engineering Manager

- Manages and maintains the CM Program.
- The Design Authority (DA) resides with the Director, Engineering and is delegated to the Piketon Engineering Manager. The DA is responsible for:
  - Establishing the design requirements
  - Ensuring design output information (documents and data) appropriately and accurately reflects the design input
- Performs and approves design/modification processes that implement the design control and design change control requirements established in the Quality Assurance Program Description (QAPD) for the American Centrifuge Plant, which includes controls for design bases, inputs, verification (including analysis software), changes, interfaces and documentation and records.
- Develops Integrated Systems and Test Plans (ISTPs).
- Manages the Temporary Change 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.

11.1.1.2.2 Director, Nuclear Safety

- Maintaining the plant's ISA, ISA Summary, and Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration.
- Identifies and defines IROFS as part of the ISA process.

11.1.1.2.3 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 Engineering organization.

11.1.1.2.4 Operations Manager

- Ensures modifications are not made to a design or operational configuration without proper review and approval.
- Assists in pre-operational tests/checks, operational, post maintenance tests/checks and post-modification tests are performed and documented to assure IROFS are operating as intended.
- Ensures work requests or other authorizations are issued prior to maintenance, testing, or modification activities.
- Ensures the occurrence of tests, calibrations, and maintenance activities are recorded.
- Ensures approved procedures are used for operations involving the replacement or adjustment of IROFS.

11.1.1.2.5 Maintenance Work Center Supervisor

- Develops and implements procedures to execute a work control process which provides for:
  - Verification of data, performance or documentation where specified by the DA; and
  - Documentation of material used to ensure design specifications are met.
- Ensures maintenance personnel are knowledgeable of requirements for working on IROFS.
- 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.

11.1.1.2.6 Training and Procedures 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 plant personnel who are relied upon to operate, maintain, inspect, or modify IROFS.
- Provides training support to engineering, operations, and maintenance personnel to ensure training is updated as a result of changes to the plant.

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 plant'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.

11.1.1.2.7 Piketon 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 DA supplied acceptance criteria are met and that accepted items are appropriately identified.
- Establishes a program for in-process inspection of maintenance work in accordance with acceptance criteria contained in maintenance procedures or provided by the DA 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.

11.1.1.2.8 Integrated Systems Test/Start-up Manager

- Assists in the development of and execution of the ISTPs which demonstrate the proper operation of completed systems to ensure that the systems meet their intended design functions.

- Ensures acceptance of turnover from the Engineering, Procurement, and Construction contractors/vendors to the Licensee, initial acceptance testing, and initial start-up of equipment and support systems.

11.1.2 Design Requirements

- Design requirements are developed to support safety functions, environmental impact-oriented functions, and mission-based functions. Defense in depth practices are applied to design, to the extent practicable. This includes the preference for engineered controls over administrative controls and minimizing challenges to IROFS.

  - IROFS are identified in the ISA Summary and Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration, with the emphasis for engineered controls over administrative controls when possible. Design requirements for IROFS or for other systems or components are required to meet the baseline design criteria (BDC) as defined in 10 CFR 70.64.

  - IROFS and other systems or components that 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 Engineering organization and documented in Design Input and Output Documents written for each system, area, and/or function. Prior to approval, these documents are reviewed to determine their adequacy, accuracy, and completeness.

- Design Input Documents provide the design basis and design requirements for the ACP. The design basis and design requirements information are found in the ISA Summary, Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration, and SRDs.

- The DA approves Design Output Documents.

- After approval by the DA, the Design Output Documents provide the baseline configuration for the plant. Drawings and specifications are examples of Design Output Documents.

- Changes to any design basis or design requirements that modify the site, structures, processes, systems, equipment, components, computer programs, or activities of personnel are controlled by the change control process described in Section 11.1.4 of this license application.

- The Design Input and Output 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 ACP 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 require generation of 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 if applicable, the commercial grade dedication requirements
- The ISA Summary, Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration, and other hazard analyses
- Procedures and plans addressing emergency operating and response plans

- Records to support maintenance and verification of the plant 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 plant, 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 ACP implements change control processes for changes to the physical plant 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 Plant Safety Review Committee reviews appropriate changes to the ACP or to ACP 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 Plant

The ACP has implemented a change control process using written procedures to control changes to the physical plant. 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 DA 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.
- Construction Project requests for plant 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;
  - ➢ Drawings/specifications; and
  - ➢ Other documentation providing a project description.
- Modifications (permanent and temporary) are evaluated, as appropriate, for any required changes or additions to the plant's procedures, personnel training, testing programs, the ISA Summary, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration. Modifications are also evaluated, as appropriate, for potential radiation exposure, potential chemical exposure, and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include: modification costs, similar completed modifications, QA aspects, potential equipment availability or maintainability concerns, constructability concerns, environmental considerations, and human factors. Modifications that establish new fissile material operations or affect existing fissile material operations are evaluated by nuclear criticality safety (NCS).
- Critical repair parts for IROFS are identified during the design process.
- Proposed plant changes 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
- 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 Sections 11.4 and 11.7 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 walk downs) are conducted periodically to confirm the adequacy of the CM function.

Initial assessments of the CM program are performed during readiness reviews of the ACP. 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 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 ACP's Corrective Action Program, described in Section 11.6 of this license application.

11.1.6 Design Verification

Many of the structures for the ACP were built by the U.S. Department of Energy (DOE) for the Gas Centrifuge Enrichment Plant program and are leased by the Licensee. Where the ACP uses existing structures, systems, or components (SSCs), the design and construction of those SSCs are verified to ensure they meet the design requirements for the ACP.

The verification process includes:

- An assessment of the SSC is conducted to compare the configuration of the SSC with original drawings, construction specifications, and procedures to the extent possible and to determine the current condition of the SSCs to the extent possible. Where appropriate, system walk-downs are performed as part of the assessment.
- The assessment results are evaluated to determine if there is a discrepancy between the installed SSC and the baseline configuration information.
- If it is determined there is a discrepancy, the necessary changes are made to correct the discrepancy.
- When it is verified that the SSC, or modified SSC, meets the design requirements, the SSC is incorporated into the baseline configuration information.

11.2 Maintenance

The Maintenance organization provides reliable and cost-effective maintenance of the ACP 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 plant 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 QA 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 Operations Manager is responsible for the overall coordination and management of the organization to provide safe and efficient performance during maintenance of plant equipment.

Maintenance Work Center Supervisor reports to the Operations Manager. The Maintenance Work Center Supervisor is responsible for directing the activities of the Balance of Plant Operations Shift Supervisors and of the Maintenance Shift Supervisors in the performance of preventive, predictive, and corrective maintenance to provide support on facilities and equipment, within approved programs, processes, procedures, and personnel training limitations. 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 Shift Supervisors, who report to the Maintenance Work Center Supervisors, are responsible for execution of maintenance on 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 equipment
- Review of work practices by craft for compliance with maintenance and plant safety procedures

Craft personnel are responsible for:

- Compliance with 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 11.3.9 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 ACP and maintenance of IROFS. Consideration is also given to the complexity of the activity, frequency 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 documents that direct performance of the maintenance activity 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 maintenance personnel. In addition, a member of the ACP organization provides oversight of contractor activities.

11.2.3 Design/Work Control

Maintenance of ACP equipment is performed in a manner that maintains the documented configuration of plant systems. Prior to modification of systems, it is necessary to complete actions required by Section 11.1 of this license application. A work control process establishes the necessary control, review, and approval process to maintain the documented configuration of ACP systems.

The need for maintenance is identified when an equipment owner initiates a request for work or by the 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. The Engineering organization classifies plant equipment to a specific QL. QLs are established in accordance with the equipment's relation to safety as determined by the ISA, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration. 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 2.0 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 QL-1 and QL-2 items. 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 QL-3 work package is required when the maintenance activity would result in a change to or creation of a quality record or 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 ACP organizations. The repair and/or replacement of IROFS are performed with like-for-like parts or substitute parts approved by the Engineering organization. Modifications to ACP systems may only be performed following evaluation and approval of the Engineering 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 ACP equipment. The Engineering organization is required to review and approve work packages created for maintenance of QL-1 and QL-2 items and packages developed for modification of ACP 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 an IROFS from service. Operations 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 activities.

The work control process provides configuration control of ACP 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; and
- Safety permits.

Other documentation related to the maintenance activity may be included in the package.

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 of this license application. Corrective actions are performed to remediate unacceptable performance deficiencies in an IROFS and to eliminate or minimize the recurrence of these deficiencies.

11.2.5 Preventive Maintenance

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. The identification, prioritization, planning, and scheduling of preventive maintenance activities are accomplished following the work control process described in Section 11.2.3 of this license application.

Establishment of a PM task is coordinated by engineering and maintenance and requires input from various disciplines within the Engineering organization, as well as operations and maintenance personnel, as appropriate. The formal documented bases for the tasks are developed, evaluated, and approved by the Engineering 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 Surveillance/Monitoring

Surveillances 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, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration 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.*

Surveillances are included in the work control process to permit timely planning, scheduling, establishment of system or plant 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 maintenance frequencies are adjusted or other corrective actions taken as appropriate.

11.2.7 Functional Testing

A post-maintenance testing (PMT) program is established to provide assurance IROFS that require a work package will perform their 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 Engineering organization may provide support to the Operations and Maintenance organizations in identifying PMT requirements. The PMT meets applicable codes and technical requirements and specifies 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 ACP equipment. These programs require 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 Engineering organization.

Additional requirements and standards are established as necessary to ensure compliance with Section 12.0 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 activity. This documentation provides information regarding the as-found condition of an IROFS. This data is used 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 Engineering organization may use 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 Qualification

The Training and Qualification program is designed to ensure that those personnel who perform activities relied on for safety have the applicable knowledge and skills necessary to design, operate, and maintain the plant in a safe manner. The Performance Based Training (PBT) methodology is used for those tasks associated with the design, modification, operation, or maintenance of IROFS identified in the ISA Summary, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration. Personnel are indoctrinated, trained and tested as necessary to ensure that they are qualified on practices important to public and worker safety, safeguarding of licensed material, and protection of the environment.

11.3.1 Organization and Management of the Training Function

The Training and Procedures Manager is responsible for establishing procedures governing the application of the PBT methodology for the analysis, design, development, implementation and evaluation of the training programs. Training personnel are assigned by the Training and Procedures Manager to interface with line managers for training development and implementation.

Instructors and subcontractors hired to develop training materials have ready access to designated subject matter experts (SMEs) who assist them when developing training materials. Training program materials are reviewed and approved by SMEs, training, and line management prior to implementation.

The functional organization managers are responsible for defining the job-specific training needs and ensuring completion of indoctrination, training, and qualification for personnel within their organization. Training attendance is tracked by training and line management. The training group 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.

Workers relied upon to design, operate, or maintain IROFS are trained and evaluated for qualifications prior to assignment of these duties. Initial training contains the classroom and on-the-job training (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. Maintenance of qualification is contingent upon successful completion of continuing training and/or through satisfactory OJT evaluations.

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.

Training materials are linked to the 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.

Training attendance records, examinations, employee qualification records, and program needs are maintained in an accurate, auditable manner to document each employee's training. The programmatic and individual training and qualification records are maintained in accordance with RMDC guidelines.

Plant functional organization managers develop and maintain a description of each individual's training requirements within their organization. These requirements are identified in individual Training Requirement Matrices (TRMs) approved by the line and training management. The TRMs include training required by regulatory and or corporate requirements in addition to the applicable Performance Based Training Requirements. Plant personnel, contractors, and visitors receive the following training as applicable to their position or function:

- **General Employee Training** for persons who require unescorted access (Section 11.3.1.1).

- **Security Education** is provided to personnel requiring plant access (Section 11.3.1.2).

- **Radiation Worker Training** for personnel whose job requires them to have unescorted access to radiological restricted areas (Section 11.3.1.3).

- **Nuclear Criticality Safety Training** for personnel who handle or manage the handling of fissile material and work within Fissile Material Operations Areas (Section 11.3.1.4).

- **Environmental, Safety, and Health Training** for those persons who have training requirements defined by laws and regulations (as defined in Section 11.3.1.5).

- **Operations and Maintenance Personnel Training** for those persons relied upon to operate or maintain IROFS. This training includes the operations and maintenance first line supervisors. (Section 11.3.1.6).

- **System Engineer Training** for those persons who review design modifications to IROFS (Section 11.3.1.7).

- **Nuclear Criticality Safety Engineer/Specialist Training** for those persons who perform the Nuclear Criticality Analyst functions described in Chapter 5.0, Nuclear Criticality Safety, of this license application (Section 11.3.1.8).

- **Health Physics Technician Training** for those persons responsible for the evaluation of radiological conditions in the plant and the implementation of the necessary radiological safety measures identified in Chapter 4.0, Radiation Protection, of this license application (Section 11.3.1.9).

- **Laboratory Technician Training [commercial ACP operations only]** for those persons who work in the laboratory technician classification (Section 11.3.1.10).

- **Fire Protection and Emergency Management Training** for those persons identified in the Emergency Plan for the American Centrifuge Plant (Section 11.3.1.11).

- **Visitor Site Access Orientation** 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

11.3.1.1 General Employee Training

General Employee Training (GET) provides awareness level training on the hazards and proper response to alarms that a person may encounter. It is required for personnel having unescorted access to the plant. GET includes the following subject areas:

- General Employee Radiological Safety
- NCS
- General Topics
- Hazard Communication
- Emergency Preparedness

11.3.1.1.1 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 as low as reasonably achievable (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 11.3.1.3 of this license application.

11.3.1.1.2 Nuclear Criticality Safety

An overview of the 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.

11.3.1.1.3 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.

11.3.1.1.4 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/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.

11.3.1.1.5 Emergency Preparedness

This training introduces personnel to basic emergency response elements including: (1) emergency plan safety objectives and priorities; (2) ways to report emergencies; (3) recognition and correct responses to plant 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.

11.3.1.2 Security Education

Security Education briefings are described in the Security Program for the American Centrifuge Plant. These include Initial Briefings, Refresher Briefings, Termination Debriefings, and Foreign Travel Briefings.

11.3.1.3 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. The extent of the course material is commensurate with the potential for exposure. The training program is reviewed and evaluated every two years.

11.3.1.4 Nuclear Criticality Safety Training

NCS training based on ANSI/ANS-8.20-1991 is provided for personnel who manage, work in, or work near the handling of fissile material and 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 plant
- NCS postings
- NCS emergency procedures
- Consequences of historical criticality accidents

Personnel are trained to report defective or anomalous NCS conditions and to perform actions only in accordance with written, approved procedures. Personnel are trained that unless a specific procedure deals with the situation, they will 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. In addition to demonstrating a basic knowledge of NCS concepts, the principles associated with the management of fissile material workers, and the oversight responsibilities of fissile material operations, NCS training for managers includes the following topics:

- Description of the plant's nuclear criticality safety policy;

- Explanation for the use of check lists, sign-off sheets, and documentation in the execution of procedures that are pertinent to criticality safety;
- Discussion of relevant procedures that pertain to criticality safety with emphasis given to criticality safety limits, controls, and emergency procedures;
- Description of the policy that relates to situations not covered by procedure and to situations in which the safety of the operation is in question; and
- Emphasizing the fact that employees are to be informed of their right to question any operation they believe may not be safe.

11.3.1.5 Environmental, Safety, and Health Training

This training covers environmental, worker safety, and health subject areas required by applicable local, state and federal regulations. It is provided to personnel commensurate with their job assignments. Specific modules identified as required compliance training for plant employees are contained in each individual's training requirement matrix. Some of the areas include:

- Radiological Worker Safety
- NCS
- 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
- Spill Prevention Control and Countermeasure Plan

11.3.1.6 Operations and Maintenance Personnel Training

Training is designed, developed, and implemented to assist plant employees in gaining an understanding of applicable fundamentals, procedures, and practices specific to the plant. 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.

11.3.1.6.1 Operations Technician

This program is designed for personnel who monitor and operate centrifuge feed, withdrawal, product, equipment and supporting systems. They operate systems necessary to support the plant, perform integrated system testing, execute valving orders, adjust equipment settings, start-up, and shutdown equipment. The Operations Technician also assemble, transfer, install, repair, and test centrifuges. 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 centrifuges.

- **Phase III** provides classroom and OJT on the IROFS identified in the ISA Summary, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration; NCS limits and controls; equipment operations; support systems; and normal, off-normal, and alarm response operating procedures for the plant.

11.3.1.6.2 Operations Shift Supervisor

This program is designed for personnel who supervise the Operations Technician 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, 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 centrifuges.

- **Phase III** provides classroom and OJT on the IROFS identified in the ISA Summary, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration; NCS limits and controls; operations; support systems; and normal, off-normal, and alarm response operating procedures for the plant.

- **Phase IV** provides classroom and OJT on the supervisory roles and responsibilities for the safe operation of the plant.

11.3.1.6.3 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 plant electrical, instrument, and electronic control systems and components.

- **Phase III** provides classroom and OJT on maintenance procedures, programs, and practices.

11.3.1.6.4 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 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 plant mechanical systems and components.

- **Phase III** provides classroom and OJT on maintenance procedures, programs, and practices.

11.3.1.6.5 Maintenance Shift Supervisor

This program is designed for the supervisors of the Maintenance 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 plant 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 plant.

11.3.1.7 System Engineer Training

System Engineer training is provided to those persons who provide engineering support; review of the design and modifications of IROFS; and review process equipment operational parameters, analyze the data and determine equipment settings. 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.

11.3.1.8 Nuclear Criticality Safety Engineer Training

Qualified Nuclear Safety personnel administer Nuclear Criticality Safety Engineer training and qualification. Training is based on ANSI/ANS-8.26-2007, *Criticality Safety Engineer Training and Qualification Program*, and ANSI/ANS-8.19-2014, *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.

11.3.1.9 Health Physics Technician Training

Health Physics support training and qualification is administered in accordance with guidelines provided in the Training Development and Administrative Guide (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 plant and the implementation of the necessary radiological safety measures as they apply to nuclear plant workers and members of the general public.

11.3.1.10 Laboratory Technician Training [commercial ACP operations only]

Laboratory support training and qualification is administered in accordance with the guidelines set down in the TDAG 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.

11.3.1.11 Fire Protection and Emergency Management Training

11.3.1.11.1 Fire Protection Training

State certification requirements provide the basis for firefighter training programs. 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 response personnel. Training needs are reviewed annually and the training program modified to meet identified needs. Drills are conducted quarterly, as part of the Emergency Plan training.

11.3.1.11.2 Emergency Management Training

Training is conducted in the areas of:

- General emergency response training
- Specialized emergency response training for the Emergency Response Organization
- Off-site Emergency Management training

Emergency Management drills and exercises are conducted to develop, maintain, and test the response capabilities of personnel, facilities, equipment, and training.

11.3.2 Analysis and Identification of Functional Areas Requiring 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, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration. The analysis is conducted with applicable program area SMEs and training personnel. The training programs for the following plant 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
- NCS Engineer
- Health Physics Technicians
- Laboratory Technicians

The plant-specific task list is developed for each of the above positions/classifications. The task lists are analyzed based on input from line management and SMEs, 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 matrices are reviewed and updated in conjunction with the periodic review of the associated procedures.

Procedure changes, equipment changes, job scope changes, plant 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 Position Training Requirements

Plant procedures and individual TRMs delineate initial and continuing training requirements for employees. The training program requirements for those positions relied on for safety or personnel who perform actions that prevent or mitigate accident sequences described in the ISA Summary, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration, are defined in TDAGs. The TDAGs include:

- Organization and Administration Responsibilities

- Trainee Selection Criteria, including the minimum educational, technical, experience, and physical requirements
- Course Loading for Initial and Continuing Training
- Test/Evaluation Guidelines
- Training and Evaluation Documentation Guidelines
- Training Courses or Modules for Specific Qualification Areas

11.3.4 Development of the Basis for Training, Including 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 from the job analysis. 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, plant systems/equipment, or job scope.

11.3.5 Organization of Instruction, Using Lesson Plans and Other 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. The lesson plans and other training guides provide the evaluation tools necessary to ensure mastery of the learning objectives.

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 plant.

Other forms of instructional materials, such as video, computer-based training and self-study may be used as alternatives or supplements to classroom instruction.

Classroom lesson plans, OJT guides, and other instructional materials receive technical reviews by designated SMEs and instructional reviews by training management as part of the approval process. The responsible line managers and Training and Procedures Manager 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.

11.3.6 Evaluation of Trainee Learning

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. Comprehensive qualification programs contain periodic evaluations of trainee performance. Remediation is provided as appropriate.

11.3.7 Conduct of On-The-Job Training

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. When the actual task cannot be performed, the conditions are documented and the task may be simulated. 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.

11.3.8 Evaluation of Training Effectiveness

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 plant training procedures.

Plant design changes, modifications, or changes in task performance are analyzed by line and training personnel for impact on 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.

Line and training management conduct self-assessments and evaluations of the individual training programs. QA auditors provide additional assessments through the audit program. These assessments and evaluations are used to determine training program strengths and weaknesses for continuous improvement of the training.

11.3.9 Personnel Qualification

Personnel are selected for entry into the training and qualification programs in conformance with the established general employment policies. The minimum education, experience, and qualification requirements for engineers, and technical professional staff, technicians, and maintenance personnel are described below. The minimum education, experience, and qualification requirements for managers and supervisors are provided in Chapter 2.0, Organization and Administration, of this license application.

Engineers and other technical professional staff, who affect the design, modification, operation, or maintenance of IROFS identified in the ISA Summary, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration, have, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and three years of nuclear experience. Other technical professional staff, whose actions are not relied upon for safety, have, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and one year of nuclear experience.

Operations technicians, maintenance personnel and technicians, and other staff whose actions are relied upon for safety 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.

Construction personnel, plant technicians, maintenance personnel, and other staff whose actions are relied upon for safety complete the applicable training programs or have equivalent experience or training.

11.3.10 Provisions for Continuing Assurance

Continuing training and periodic requalification 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 and is 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, plant or system changes, component changes, procedure changes, lessons learned (including industry and in-house operating experiences, and event reports), and emergency response procedures.

11.3.11 References

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

2. ANSI/ANS-8.19-2014, *Administrative Practices for Nuclear Criticality Safety*

3. ANSI/ANS-8.26-2007, *Criticality Safety Engineer Training and Qualification Program*

11.4 Procedures

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, plant employees, and the environment. Procedures are used to ensure safe work practices and apply to workers, visitors, contractors, and vendors. 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 plant operation, meet regulations and standards, and assure a balance of effective safety with practical efficiency in plant operations. Activities involving nuclear material and/or IROFS are conducted in accordance with approved procedures.

A management controls program for procedures includes the basic elements of identification, development, verification, review and comment resolution, approval, validation, issuance, and change control, and periodic review. These elements are outlined in a procedures management writer's guide and described in implementing procedures.

11.4.1 Types of Procedures

Procedures are intended to prescribe those essential actions or steps needed to safely and consistently perform operations and maintenance activities. Procedures that are related to the operation of IROFS where human actions are important and for the management measures supporting those IROFS are governed by the requirements of this section. The two general types of procedures used at the ACP are Operating and Administrative.

11.4.1.1 Operating Procedures

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
- Steps for each operating process phase
- Initial start-up
- Normal operations
- Temporary operations
- Emergency shutdown
- Emergency operations
- Normal shutdown
- Start-up following an emergency or extended downtime
- 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
- 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

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 affected parties (technicians and supervisors) before performing work and on completion of maintenance work. The discussion includes potential degradation of IROFS during the planned maintenance.

Alarm Response Procedures 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 describe actions to be taken during unusual or out-of-the ordinary situations.

Emergency Operating Procedures direct actions necessary to mitigate potential events or events in progress that involve needed protection of on-site personnel; public health and safety; and the environment.

11.4.1.2 Administrative or Management Control Procedures

Administrative procedures or "management control procedures" are used for activities that support the process operations. These procedures are used to manage activities such as configuration management, radiation protection, maintenance, QA, training and qualification, audits and assessments, incident investigations, record keeping, and reporting. Administrative procedures direct the following activities:

- Design
- Configuration Management
- Procurement
- Construction
- Radiation safety
- Maintenance
- QA elements
- Training and qualification
- Audits and assessments
- Incident investigations

- Records management
- Criticality safety
- Fire safety
- Chemical process safety and reporting requirements

11.4.2 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 ensures that:

- Procedures are identified and developed as needed;
- Procedures are provided for those operations of IROFS where human actions are necessary and for the Management Measures described in this chapter;
- Essential elements that are generic are included as applicable. These include: nuclear criticality; chemical process and fire safety; warnings and cautions; notes or reminders of pertinent information regarding specific hazards or concerns; Material Safety Data Sheet/Safety Data Sheet availability; special precautions; radiation and explosive hazards; and special personal protective equipment;
- Procedures are approved under the guidelines of the configuration management program by personnel responsible and accountable for the operation;
- Procedures are verified and validated through field tests by workers and technicians during procedure development to provide assurance that they are usable and accurate;
- Procedures are periodically reviewed and re-verified and validated;
- Current procedures are available to personnel and that users are qualified on the latest version;
- Operating limits and IROFS are specified in the procedure;
- Safety limits and IROFS will be clearly identified, as such, in the procedure for operations;
- Procedures include required actions for off-normal conditions of operation, as well as normal operations;
- If needed, hold points or safety checkpoints are identified at appropriate steps in the procedure;

- A mechanism is specified for revising and reissuing procedures in a controlled manner;
- Current procedures are available and used at work locations; and
- The plant Training Program trains the required persons in the use of the latest procedures available.

The procedure process utilizes nine basic elements to accomplish procedure development, review, approval, and control: Identification; Development; Verification; Validation; Review and Comment Resolution; Approval; Issuance; Change Control; and Periodic Review. These elements are discussed in the following sections.

11.4.2.1 Identification

ACP organization managers have the responsibility for identifying which tasks will be proceduralized within their areas of control, using the criteria in the following paragraphs below and Section 11.4.9 of this license application.

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 and Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration
- Operator actions necessary to prevent or mitigate the consequences of accidents described in the ISA Summary and Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration
- Safe work practices to control processes and operations with special nuclear material, IROFS, and/or hazardous chemicals incident to the processing of licensed material.

A detailed procedure is normally not needed if the task analysis determines that:

- 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

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*.

11.4.2.2 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.
- The following elements will be considered for procedure incorporation:
  - Title and identifying information, such as number, revision, and date
  - Statement of applicability and purpose
  - Prerequisites
  - Precautions (including warnings, cautions, and notes)
  - Important human actions
  - Limitations and actions
  - Acceptance criteria
  - Checkoff lists
  - Reference material
- Interviews with procedure users and process walk downs 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 11.4.7 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.
- 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 11.4.2.6 of this license application.

11.4.2.3 Verification

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 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.

11.4.2.4 Validation

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.

11.4.2.5 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. Nuclear criticality safety reviews drafts of new procedures and procedure changes that could affect fissile material operations.

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).

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; such as an accident, unexpected transient, significant operator error, or equipment malfunction to determine if changes are appropriate based on the cause and corrective action determination for the particular incident. Procedure changes that are necessary because of a system modification are addressed in Section 11.1 of this license application, as part of the modification control process.

In addition, the Plant Safety Review Committee will review:

- Each new procedure required by Section 11.4.2.1 for this license application

- Each proposed change to procedures required by Section 11.4.2.1 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)

11.4.2.6 Approval

Following the resolution of review comments, procedures are approved. Approval authority rests with the applicable ACP organization manager responsible for the activity.

Managers ensure that appropriate training is completed on new and revised procedures.

11.4.2.7 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.

11.4.3 Procedure Hierarchy

The procedure hierarchy is established in four levels. The levels are:

- **Level 1** - Policy statements issued by executive management that apply to ACP 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

11.4.4 Temporary Changes

Temporary changes to procedures required by Section 11.4.2.1 of this license application can be made, provided:

- The temporary change does not result in a change to the ISA, or Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration 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 exists 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 11.4.2.4 of this license application.

11.4.5 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 building, 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 this period are assessed to ensure it is appropriate to extend the use of the temporary procedure or to develop a permanent procedure. These temporary procedures are subject to the same level of review and approval as required for permanent procedures.

11.4.6 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 SME performs a complete administrative and technical (requirements and references) review ensuring information is complete and accurate and that the procedure is usable as written.

11.4.7 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 (e.g., process flow-charts or component identification tables) 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.

If a step of a procedure cannot be performed as written, work is stopped, the system is immediately placed in a safe condition, and corrective actions are initiated in accordance with plant 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.

11.4.8 Records

Records generated during procedure use are identified in the governing procedure and controlled according to the ACP RMDC program practices as described in Section 11.7 of this license application.

11.4.9 Topics to be Covered in Procedures

Activities defined by Section 11.4.2.1 of this license application are the minimum activities that are to be covered by written procedures. In addition, any activity described in Section 11.4.2.1 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 plant 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 safety or 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
  - Facility utilities (for example: air, nitrogen, cooling water, sanitary water, site water)
  - Temporary changes in operating procedures
  - Purge and evacuation vacuum systems
  - Installation and removal of centrifuges
- **ABNORMAL OPERATION/ALARM RESPONSE**
  - Loss of cooling
  - Loss of instrument air
  - Loss of electrical power
  - Fires
  - Chemical process releases

- ➢ Loss of feed capacity
- ➢ Loss of withdrawal capacity
- ➢ Loss of purge vacuum

- **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
  - ➢ Centrifuge assembly/installation

- **EMERGENCY PROCEDURES**
  - ➢ Toxic chemical releases (including UF_6)

11.4.10 References

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

2. LA-3605-0003, In*tegrated Safety Analysis Summary for the American Centrifuge Plant*

3. LA-3605-0003A, *Addendum 1 of the* Int*egrated Safety Analysis Summary for the American Centrifuge Plant - HALEU Demonstration*

11.5 Audits and Assessments

The ACP 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 Piketon QA organization in accordance with written procedures or checklists by qualified auditors. The auditors are independent from activities being audited. 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 the plant operations are being conducted safely in accordance with regulatory requirements, license application commitments, and the ISA.

These audits and their associated frequencies are conducted in accordance with Section 18.0 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 Engineers (ASME) NQA-1-2008, Part 1, Requirement 2, Section 300, *Qualification Requirements*, and Section 400, *Records of Qualification.* 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 plant and/or external technical specialists.

In addition to periodically evaluating aspects of the QAPD, audits are conducted for the areas of radiation safety; NCS [every two years]; nuclear safety; chemical safety; fire safety; environmental protection; emergency management; QA; CM, maintenance; training and qualification; procedures; incident investigation; records management; security (every two years); and operations.

Audit results are documented and reported to the plant senior management as specified in plant procedures. Provisions are made for reporting and corrective action, where warranted. The plant 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 16.0 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, NCS; nuclear safety; chemical safety; fire safety; environmental protection; emergency management; QA; CM; maintenance; training and qualification; procedures; incident investigation; records management; and operations. Operational assessments will also be performed to ensure the operational assumptions as defined in the ISA Summary and Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration are valid.

Assessment results are documented and reported as specified in the plant procedures. Provisions are made for reporting and corrective action, where warranted, in accordance with the plant's Corrective Action Program.

Additional requirements for performing Nuclear Criticality Safety Assessments are specified in Chapter 5.0 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 ACP. 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 ACP is required by 10 CFR 70.50 and 70.74 to notify the NRC of certain 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 ACP 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 ACP are promptly reported to appropriate personnel, assessed, and when required, reported to the NRC Operations Center or designated NRC office.

11.6.1 Incident Identification, Categorization, and Notification

In accordance with procedures, plant personnel are required to report to their line manager or directly to the Operations Shift Supervisors and/or Plant Shift Superintendent (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.

The Operations Shift Supervisors and/or 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 Operations Shift Supervisor may consult with ACP senior management or other personnel possessing expertise or knowledge concerning the type of event or condition being assessed.

If an event or condition within the plant 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 30 minutes and 24 hours. Verbal and/or written communication involving classified information is conducted in accordance with the SP-3605-0041, *Security Plan for the Protection of Classified Matter at the American Centrifuge Plant.*

11.6.2 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 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. The investigator(s) are independent from the line function involved with the incident under investigation. A procedure provides a documented plan for investigating abnormal events and includes the functions, responsibilities, and scope of authority of investigators. 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. 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.

The investigator(s) have the authority to obtain all the information considered necessary during the course of the investigation and participants of an investigation team are assured of no retaliation for participation in an investigation. Line management cooperates fully with the investigators. 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.

11.6.3 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.

The 10 CFR 70.50(c)(2) reporting criteria require that the ACP submit a written follow-up report within 30 days of the initial report required by 10 CFR 70.50 (a) or (b) or by 10 CFR 70.74 and Appendix A of Part 70. In lieu of the 30-day requirement described in 10 CFR 70.50(c)(2), NRC approval to submit the required written reports within 60 days of the initial notifications is hereby requested. This exemption request is provided in Section 1.2.5 of this license application.

11.6.4 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. Evidence files used to support action closure are maintained in accordance with approved records management procedures.

Documentation is maintained so that "lessons learned" may be applied to future operations of the ACP. Details of the event sequence are compared with accident sequences already considered in the ISA. Should it be necessary, the ISA Summary and Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration are modified to include evaluation of the risk associated with accidents of the type actually experienced. Relevant findings from incident investigations are reviewed with affected ACP personnel.

The Corrective Action Program also requires that initiating events, as defined in the ISA Summary and Addendum 1 of the ISA Summary for the ACP – HALEU Demonstration, will be reviewed and tracked to ensure that the frequency with which they occur does not exceed the assumptions made in the ISA. Should those reviews indicate that the frequencies are not conservative, appropriate actions will be taken to ensure the 10 CFR 70.61 Performance Requirements are met.

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 10 CFR 70.64(a) and the defense in depth requirements of 10 CFR 70.64(b), and the requirements contained in the Quality Assurance Program Description (QAPD). 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 and check lists must include 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 designee, as specified in the procedure, which controls the generation and revision of these records.

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 by the manager, or designee, of the organization that originates the record. Lifetime records are retained for the life of the item to which they apply or as required by a regulatory agency. 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. Lost or damaged records are replaced, unless deemed impractical with the concurrence of the QA organization. Single copy records are checked out of storage only if they cannot be copied and then only for a limited period. Temporary protection in such cases is provided by prudent business practices (e.g., record of custody, office environment, and workplace security). 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, rodents and dust or airborne particles. The applicable document that specifies the record indicates those to be forwarded for lifetime storage. Storage facilities are maintained to ensure continuous protection of the records. Requirements are specified for both permanent and temporary storage of records.

- **Permanent Storage**

  - Single storage consist of facility, vault, room or container with a minimum 2-hour fire rating. The design and construction of a single storage facility, vault room, or container shall be reviewed for adequacy by a person competent in fire protection or contain a certification or rating from an accredited organization.; or

  - Dual facilities, containers or a combination thereof that are sufficiently remote from each other to eliminate the possibility of exposure to simultaneous hazards.

- **Temporary Storage**

  The RMDC process requires that those completed records documenting nuclear safety or safeguards and security matters, which are being held temporarily by originating organizations, be properly protected by maintaining them in 1-hour, fire-rated facility or container. If 1-hour fire-rated container is used they either bear an Underwriters Laboratory label (or equivalent) certifying 1-hour fire protection, or the container is 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 and sensitive records are in accordance with SP-3605-0041, *Security Plan for the Protection of Classified Matter at 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 plant 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 of this license application. The records are listed under the chapter headings of the NUREG-1520, *Standard Review Plan for Fuel Cycle Facilities License Applications*. 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 and 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 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 Items Relied On For Safety Failures

Records of IROFS failures are kept and updated in accordance with 10 CFR 70.62 (a)(3). Record revisions necessitated by post-failure investigation conclusions will be made promptly in accordance with 10 CFR 70.62(a)(3) based on the nature of the record, extent of revision necessary, and potential safety significance. Necessary record revisions will be made within 30 days of the completion of the investigation, unless specifically approved by ACP management

11.7.1.16 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 the procedures described in 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 verified adequate, approved and released for distribution. Organizations that are authorized to approve and distribute controlled documents are identified in the plant 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 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 copyholders. Copyholders 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 copyholders 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 or sensitive documents is accomplished in accordance with SP-3605-0041, *Security Plan for the Protection of Classified Matter at 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 copyholders are defined. Responsibilities include requirements for the use of controlled documents and working copies. Copyholders 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 and 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 and data as hardware and software technology change. For example, procedures allow older forms of information and codes for older computing equipment to be transferred to contemporary computing media and equipment. Routine backups of the Document Control database are performed by application administrators.

11.7.2.13 Assessment

The overall effectiveness of the Document Control program is evaluated through the audit program described in Section 18.0 of the QAPD. Deficiencies identified are corrected in a timely manner in accordance with the requirements described in Section 11.6 of this license application.

11.7.2.14 Archiving Documents

Revisions of controlled documents are transmitted to RMDC and the previous revision of the document is archived in accordance with the requirements of the Document Control Process.

11.7.3 Organization and Administration

11.7.3.1 Responsibilities

The Training and Procedures Manager is responsible for the RMDC program. These responsibilities include:

- Directing the 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 the 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

The following are examples of the types of records maintained by RMDC.

- **Chapter 1.0 - General Information**
  - Construction records
  - Plant and equipment descriptions and drawings
  - Design criteria, requirements, and bases for IROFS as specified by the ACP CM function
  - Records of plant changes and associated integrated safety analyses, as specified by the ACP CM function
  - Safety analyses, reports, and assessments
  - Records of site characterization measurements and data
  - Records pertaining to on-site 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.
  - QA 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
  - NCS evaluations
  - Records pertaining to nuclear criticality inspections, audits, investigations, and assessments
  - Records pertaining to nuclear criticality incidents, unusual occurrences, or accidents
  - Records pertaining to NCS evaluations
- **Chapter 6.0 - Chemical Safety**
  - Chemical process safety procedures and plans
  - Records pertaining to chemical process inspections, audits, investigations, and assessments
  - Chemical process 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

    - Training procedures

    - Training modules

  - 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 QA Elements

- ❖ Inspection records
- ❖ Test records
- ❖ Corrective action records

11.8 Other Quality Assurance Elements

The plant has developed QA principles as described in Section 1.0 of the QAPD.

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