Orano Enrichment USA, LLC

Enrichment Facility License Application

Revision 0

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List of Acronyms

Acronym	Definition
AB	Administration Building
AEGL	Acute Exposure Guideline Level
ALARA	As Low As Reasonably Achievable
ALI	annual limit on intake
AMSL	Above mean sea level
ANSI	American National Standards Institute
APF	Assigned Protection Factor
BDC	Baseline Design Criteria
BSPB	Blending, Sampling and Preparation Building
CAA	Controlled Access Area
CAAS	Criticality Accident Alarm System
CAB	Centrifuge Assembly Building
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CI	Configuration Items
CRSB	Cylinder Receiving & Shipping Building
DAC	derived air concentration
DCE	Decommissioning Cost Estimate
DFP	Decommissioning Funding Plan
DP	Decommissioning Plan
DOE	U.S. Department of Energy
DTF	Decommissioning Trust Fund
DUF_6	Depleted Uranium
EECP	Entry Exit Control Point
EFPC	East Fork Poplar Creek
EHS&L	Environmental, Health, Safety and Licensing

Acronym	Definition
EPA	U.S. Environmental Protection Agency
EPC	Engineering, Procurement and Construction
ER	Environmental Report
ERO	Emergency Response Organizations
ERPG	Emergency Response Planning Guideline
ESB	Electrical Service Buildings
ETCSP	Empty / Tails Cylinder Storage Pad
FA	Function Allocation
FB	Fire Brigade
FHA	Fire Hazards Analysis
FNMC	Fundamental Nuclear Material Control plan
FPCSP	Full Product Cylinder Storage Pad
FOCI	Foreign Ownership, Control and Influence
FRA	Functional Requirements Analysis
GDFS	Gasoline and Diesel Fueling Station
GEV	Gaseous Effluent Ventilation
HEPA	High Efficiency Particulate Air
HF	Hydrogen Fluoride
HFE	Human Factors Engineering
HSI	Human System Interface
HVAC	Heating, Ventilation and Air Conditioning
ICRP	International Commission on Radiological Protection
I&C	Instrumentation and Control
IROFS	Items relied on for safety
ISA	Integrated Safety Analysis
ISAS	Integrated Safety Analysis Summary
LES	Louisiana Enrichment Services
LFACP	local fire alarm control panel

Acronym	Definition
MC&A	Nuclear Material Control & Accounting
MFACP	main fire alarm control panel
MT	Metric Tons
NCRP	National Council on Radiation Protection and Measurements
NCS	Nuclear Criticality Safety
NCSA	Nuclear Criticality safety analyses
NCSE	Nuclear Criticality safety evaluations
NFPA	National Fire Protection Association
NIOSH	National Institute of Occupational Safety and Health
NPH	natural phenomena hazards
NRC	U.S. Nuclear Regulatory Commission
NSI	National Security Information
OE	Orano Enrichment USA LLC
OER	Operating Experience Review
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
OSB	Operation Support Building
OSHA	Occupational Health and Safety Administration
PEC	Passive Engineered Control
PFPE	perfluorinated polyether
PM	Preventative Maintenance
QA	Quality Assurance
QAPD	Quality Assurance Program Description
QC	Quality Control
RD	Restricted Data
REMP	Radiological Environmental Monitoring Program
ROI	Region of Influence
RWS	Radiation Work Permit

Acronym	Definition
SAR	Safety Analysis Report
SBM	Separations Building Modules
SGI	Safeguards Information
SM	Source Material
SNM	Special Nuclear Material
SRC	Safety Review Committee
SSC	Structure, Systems, and Components
SSP-2	Self Sufficiency Parcel – 2
SWU	Separative Work Unit (of enrichment)
TA	Task Analysis
TEDE	Total Effective Dose Equivalent
TLDs	thermoluminescent dosimeters
TSB	Technical Support Building
TVA	Tennessee Valley Authority
U	Soluble Uranium
UF_4	Uranium Tetrafluoride
UF_6	Uranium Hexafluoride
UO_2F_2	Uranyl Fluoride
UPS	Uninterruptible Power Supply
USL	Upper Safety Limit
$^{w}/_{o}$	Weight percent

Chapter 1 – General Information

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1.0 GENERAL INFORMATION

This section contains a general description and purpose of the Orano Enrichment USA LLC (OE) uranium enrichment facility, hereafter referred to as the "Facility". The Facility enriches uranium for the production of nuclear fuel for use in commercial power plants. This License Application follows the format recommended by NUREG-1520 (NRC 2015). The level of detail provided in this chapter is appropriate for general familiarization and understanding of the facility and processes. The information is to be used as background for the more detailed descriptions provided in other chapters of the license application, the Environmental Report (ER), or the Integrated Safety Analysis Summary (ISAS). This chapter also provides information on the corporate structure and economic qualifications of OE.

This License Application is pursuant to 10 CFR Parts 30, 40 and 70, authorizing OE to receive, acquire, possess, and transfer byproduct, source, and special nuclear material.

1.1 FACILITY AND PROCESS OVERVIEW

The design of the Facility incorporates the latest safety improvements and design enhancements from the enrichment facilities currently operating and under construction in Europe.

The primary function of the Facility is to enrich natural uranium hexafluoride (UF_6) by separating a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream enriched in ^{235}U and a tails stream depleted in the ^{235}U isotope. The feed material for the enrichment process is UF_6 with a natural composition of isotopes ^{234}U, ^{235}U, and ^{238}U. The enrichment process is a mechanical separation of isotopes using a fast rotating cylinder (centrifuge) based on a difference in centrifugal forces due to differences in molecular weight of the uranic isotopes. No chemical changes or nuclear reactions take place. The feed, product, and tails streams are all in the form of UF_6.

1.1.1 Facility Layout Description

Site features are well suited for the location of a uranium enrichment facility as evidenced by its favorable conditions of hydrology, geology, seismology, and meteorology as well as good transportation routes for transporting feed and product by truck.

The Facility is located on approximately 200 acres of approximately 600 acres of the SSP-2 Parcel on the Oak Ridge Reservation (ORR) in Oak Ridge, Tennessee. The Separations Building Modules, Administration Building, Security and Secure Administration Building, Cylinder Receipt and Shipping Building, Centrifuge Assembly Building, Electrical Services Building, Mechanical Services Buildings, Technical Support Building, Operation Support Building, and Cylinder Storage Pads are located in the western portion of the Site on about 100 acres of developed area.

A layout of the facility on SSP-2 is shown in Figure 1-1.

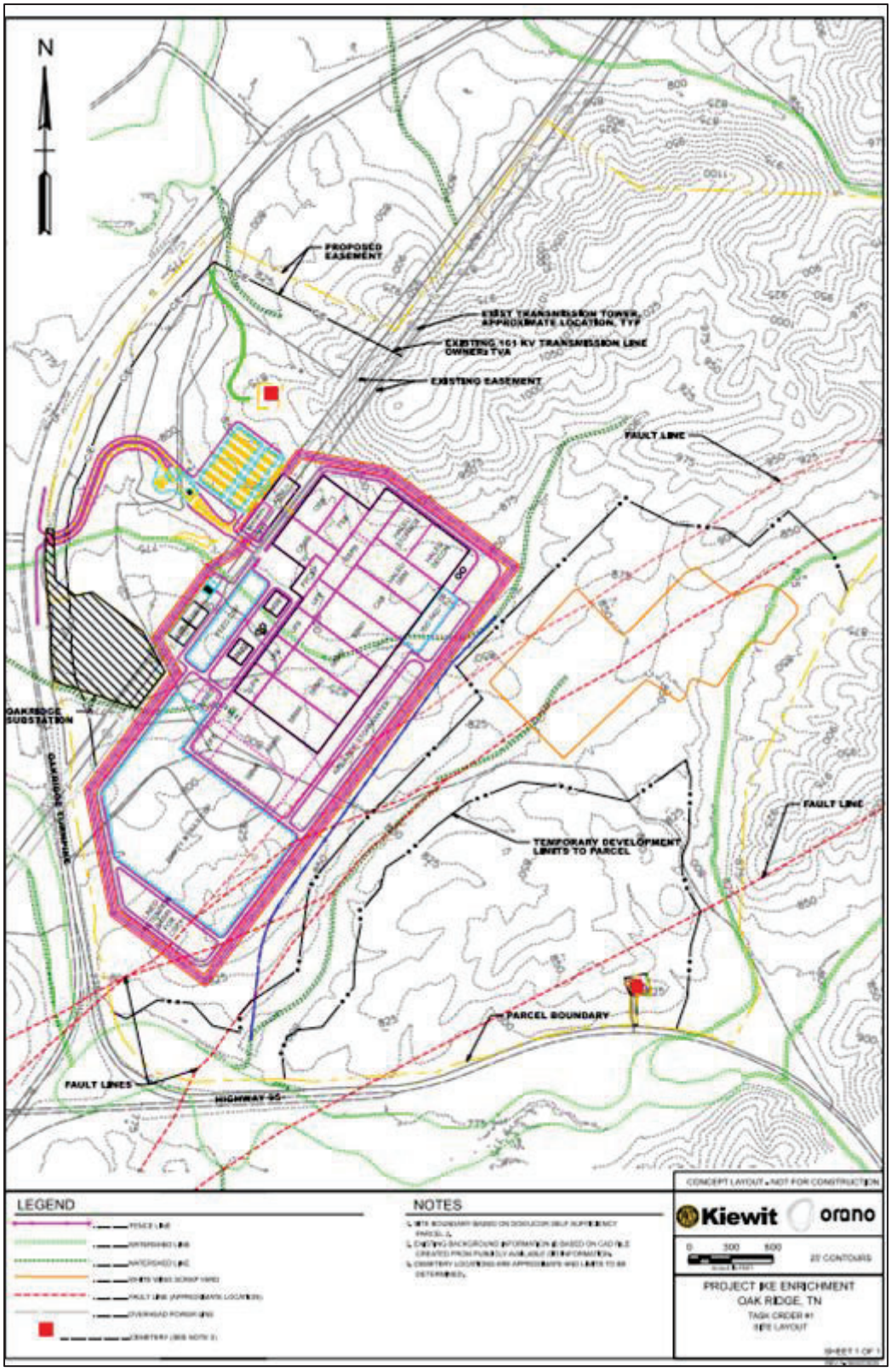

Figure 1-1 Facility Layout on SSP-2

1.1.1.1 Guard House

The main Facility Guard House is located at the entrance to the Facility. It functions as needed as a security checkpoint for all incoming and outgoing traffic. Employees, visitors and trucks that have access approval will be screened at the main Guard House. A smaller Vehicle Inspection Guard House, an extension of the Security and Secure Administration Area is also located the vehicle access point into the Controlled Access Area of the facility.

1.1.1.2 Administration Building

The Administration Building (AB) is on the northwest side of the site. It contains general office areas and a cafeteria. The AB is adjacent to the main parking lot. Vehicular traffic passes through a security checkpoint before being allowed to park. Parking is located outside of the Controlled Access Area security fence. Personnel enter the AB and general office areas via the main lobby and interior security checkpoint.

1.1.1.2.1 Security and Secure Administration Area

The Security and Secure Administration Area is in the AB and contains secure office areas and the Facility's Entry Exit Control Point (EECP). The EECP is designed to facilitate and control the passage of authorized facility personnel and visitors.

1.1.1.3 Centrifuge Assembly Building

This building is used to assemble centrifuges before they are moved into the Separations Building and installed in the cascades. Source material and SNM are used and produced in this area.

The major functional areas of the Centrifuge Assembly Building are:

- Centrifuge Component Storage Areas
- Centrifuge Assembly Areas
- Assembled Centrifuge Storage Areas
- Building Office Area
- Centrifuge Test Facility
- Post-Mortem Centrifuge Facility

1.1.1.4 Cylinder Receipt and Shipping Building

The Cylinder Receipt and Shipping Building (CRSB) is located near the Cylinder Storage Pads. This building contains equipment to receive, inspect, weigh and temporarily store cylinders of feed UF_6 sent to the Facility; temporarily store, inspect, weigh, and ship cylinders of enriched UF_6 to facility customers; receive, inspect, weigh, and temporarily store empty product and DUF_6 cylinders prior to being filled in the Separations Building Module (SBM); and inspect, weigh, and transfer filled DUF_6 cylinders to the Empty / Tails Cylinder Storage Pad (ETCSP). Source material and SNM are used in this area.

The functions of the CRSB are:

- Loading and unloading of cylinders
- Weighing and inspecting arriving feed cylinders
- Preparation of cylinder overpack protective packaging, as required

Source material and SNM are used in this area.

1.1.1.5 Technical Support Building

The Technical Support Building (TSB) contains radiological support areas for the Facility. It also acts as a secure point of entry to the SBMs and the Blending, Sampling and Preparation Building (BSPB). Source material and SNM are found in this area. The TSB and the Operation Support Building (OSB) are separate, adjacent areas of the same physical structure.

The major functional areas of the TSB are:

- Solid Waste Collection Room
- Decontamination Workshop
- Liquid Effluent Collection and Treatment Room
- TSB/BSPB Gaseous Effluent Ventilation System
- Laboratory Areas
- Radiation Monitoring Room
- Maintenance Facility for contaminated facility equipment
- Ancillary Areas

1.1.1.6 Operation Support Building

The OSB is an area within the same physical building as the TSB and is adjacent to the CRSB. The OSB contains non-radiological support areas for the Facility. The OSB forms the entrance to the TSB and is directly across the street from the AB.

The OSB contains the following functional areas:

- Control Room
- Medical Room
- Locker Rooms
- Break Area
- Security Lobby
- Security Alarm System Room
- Ancillary Areas

1.1.1.7 Blending, Sampling and Preparation Building

The BSPB is adjacent to the UF_6 Annexes, the CRSB, and the TSB.

The primary function of the BSPB is to provide means to:

- Blend cylinders with different enrichment level at the required ^{235}U enrichment level - Product Blending and Transfer System
- Liquefy, homogenize and sample cylinders prior to shipment to the customer - Product Liquid Sampling System
- Testing, weighing, conditioning, defrosting, inspection, and preparation of cylinders

Source material and SNM are used in this area.

1.1.1.8 Separation Building Modules

The Facility includes four identical SBMs. Each SBM consists of two Modules. Each Module has twelve cascades with each cascade having hundreds of centrifuges. Each Module is capable of producing approximately 925,000 SWU per year. Source material and special nuclear material (SNM) are used or produced in this area.

The major functional areas of the SBM are:

- Modules (2)
- Process Service Corridor
- UF_6 Annexes providing:
  - Feed and purification stations
  - Product and tails take off stations
  - Gaseous effluent treatment
  - Mass spectrometry sampling of feed, product, and tails flows
  - Centrifuge power conditioning

Source material and SNM are used or produced in this area.

1.1.1.9 Cylinder Storage Pads

The Facility uses several outside areas for the storage of full cylinders containing UF_6 and for storage of empty cylinders. Cylinders containing UF_6 that is depleted in ^{235}U are stored on the ETCSP. Full feed cylinders containing natural UF_6 are stored on the Feed Cylinder Storage Pad (FCSP). Empty cylinders are stored on the ETCSP. The ETCSP is at the southern end of the Facility and consists of multiple adjacent pads where cylinder types are arranged to provide the maximum flexibility from an operations perspective while still minimizing dose to operators from recently emptied cylinders. Full product cylinders containing enriched UF_6 are stored on the Full Product Cylinder Storage Pad (FPCSP). Empty product cylinders containing UF_6 heels are stored on the FPCSP. The FPCSP is located near the BSPB adjacent to the CRSB.

1.1.1.10 Electrical Services Buildings

The Electrical Services Buildings are located immediately west of the SBMs. They house standby diesel generators, which provide the site with standby power.

1.1.1.11 Gasoline and Diesel Fueling Station

A Gasoline and Diesel Fueling Station (GDFS) is located to the west of the SBM. The GDFS supports vehicle fueling from an adjacent fuel pump island and on-site vehicle minor repair and maintenance conducted inside the building.

1.1.1.12 Mechanical Services Buildings

The two Mechanical Service Buildings are located west of the SBMs. They house air compressors, the demineralized water system, the centrifuge cooling water system pumps, heat exchangers, and expansion tanks.

1.1.2 Process Overview

This section provides a description of the various processes analyzed as part of the Integrated Safety Analysis (ISA). A brief overview of the entire enrichment process is provided followed by an overview of each major process system. A detailed description of these processes and the systems that provide them is found in the ISAS.

1.1.2.1 Process Overview

The Facility is based on a highly reliable gas centrifuge process. The Facility is designed to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream - enriched in the uranium-235 (^{235}U) isotope and a tails stream - depleted in the ^{235}U isotope (DUF_6). The process, entirely physical in nature, takes advantage of the tendency of materials of differing density to segregate in the force field produced by a centrifuge. The chemical form of the working material of the plant, UF_6, does not require chemical transformations at any stage of the process. This process enriches natural UF_6, containing 0.711% ^{235}U to a UF_6 product, containing up to 10% of ^{235}U.

Separation operations are divided among four SBMs. Each SBM is divided into two Modules, each containing 12 centrifuge cascades that produces enriched UF_6 at a specified enrichment level, so up to eight different enrichment levels can be produced at one time. Enrichment support systems are dispatched in the BSPB and in the TSB.

1.1.2.2 Process System Descriptions

An overview of the enrichment process systems and the enrichment support systems is discussed below.

Numerous substances associated with the enrichment process could pose hazards if they were released into the environment. Chapter 6 contains a discussion of the criteria and identification of the chemicals of concern at the Facility and concludes that UF_6 is the only chemical of concern that will be used at the Facility.

For additional information on all Process System descriptions, refer to Section 2 of the ISAS.

1.1.2.2.1 UF_6 Feed System

The first step in the process is the receipt, inspection and weighing of the 48-inch feed cylinders in the CRSB before transfer to the UF_6 Feed System located in the UF_6 Annex of the SBM.

The function of the UF_6 Feed System is to provide a continuous supply of gaseous UF_6 from the feed cylinders to the cascades.

Feed cylinders are loaded into Feed Donor Stations and heated to sublime the UF_6. Before UF_6 transfer to the cascades, light gases, primarily air and hydrogen fluoride (HF) present in the UF_6 feed are routed to the Feed Purification Subsystem. Purification cylinder will be reused as feed cylinder after light gas purging and analysis.

1.1.2.2.2 Cascade System

Multiple gas centrifuges make up arrays called cascades. The Cascade System separates gaseous UF_6 feed with a natural uranium isotopic concentration into two process flow streams - product and tails. The product stream is ^{235}U enriched up to 10%. The tails stream is UF_6 that has been depleted of ^{235}U isotope below 0.4 % ^{235}U.

1.1.2.2.3 Product Take-off System

The function of the Product Take-off System is to provide continuous withdrawal of the enriched gaseous UF_6 product from the cascades and to purge and dispose of light gas impurities from the enrichment process.

The product stream leaving the cascades is transported at subatmospheric pressure via a train of vacuum pumps to Product Receiver Stations in the UF_6 Annex into a 30-inch cylinder under chilled air to effect the UF_6 desublimation.

The Product Take-off System also contains a system to purge light gases (typically air and HF) that may have passed through the centrifuges. This system consists of UF_6 Cold Traps which capture UF_6 while leaving the light gas in a gaseous state. The cold trap is followed by product vent Vacuum Pump/Trap Sets, each consisting of a sodium fluoride trap, an alumina trap, and a vacuum pump. The sodium fluoride trap removes small traces of UF_6 and the alumina trap removes any HF from the product gas.

Any UF_6 captured in the cold trap is periodically transferred to another product cylinder for use as product or blending stock. Filling of the product cylinders is monitored with a load cell system, and filled cylinders are transferred to the Product Liquid Sampling System for sampling.

1.1.2.2.4 Tails Take-off System

The primary function of the Tails Take-off System is to provide continuous withdrawal of the gaseous UF_6 tails from the cascades. A secondary function of this system is to provide a means for removal of UF_6 from the centrifuge cascades under abnormal conditions.

The tails stream exits each Module via a primary header, goes through a pumping train, and then to Tails Receiver Stations in the UF_6 Annex. Chilled air flows over cylinders in the Tails Receiver Stations to effect the desublimation. Filling of the cylinders is monitored with a load cell system, and filled cylinders are transferred outdoors to the ETCSP.

1.1.2.2.5 Product Blending and Transfer System

The primary function of the Product Blending and Transfer System is to provide a means to fill 30-inch cylinders with UF_6 at a specific enrichment of ^{235}U to meet customer requirements. This is accomplished by blending (mixing) UF_6 at two different enrichment levels to one specific enrichment level. The system can also be used to transfer product from a 30-inch or 48-inch cylinder to another 30-inch cylinder without blending. Additionally, the Product Blending and Transfer System can be used to evacuate light gas in full, partially full, and empty cylinders and reduce the heel quantities in cylinders.

This system consists of Blending Donor Stations (which are similar to the Feed Donor Stations) and Blending Receiver Stations (which are similar to the Product Receiver Stations) described under the primary systems.

1.1.2.2.6 Product Liquid Sampling System

The function of the Product Liquid Sampling System is to obtain an assay sample from all product 30-inch cylinders. The samples are used to measure whether the enriched UF_6 meets the chemical, physical, and isotopic composition requirements specified in ASTM C996 for customer acceptance (ASTM 2025). Sampling of 48-inch cylinders filled for internal use (feed, purification or product) is also conducted through this system.

1.1.2.2.7 Cylinder Preparation System

The Cylinder Preparation process includes the performance of certain tests and inspections on full or partially full cylinders and cylinders containing heels, and testing and replacing cylinder valves.

1.1.2.2.8 Support Systems

Support functions, including chemistry analysis, equipment decontamination and maintenance, liquid effluent treatment are conducted in the TSB. Decontamination, primarily of pumps and valves, uses solutions of citric acid. Liquid effluent is collected and treated using the Liquid Effluent Treatment System. There are no liquid discharges to the environment from this system.

1.1.2.2.9 Waste Management

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

Refer to Section 1.1.4 for additional information.

1.1.3 Site Overview

The Facility is located near the city of Oak Ridge, Tennessee. The Facility will be located on an approximate 600 acres site which will be acquired from the Industrial Development Board of Oak Ridge (the Site).

The geographic location of the facility is shown on Figure 1-2 and Figure 1-3.

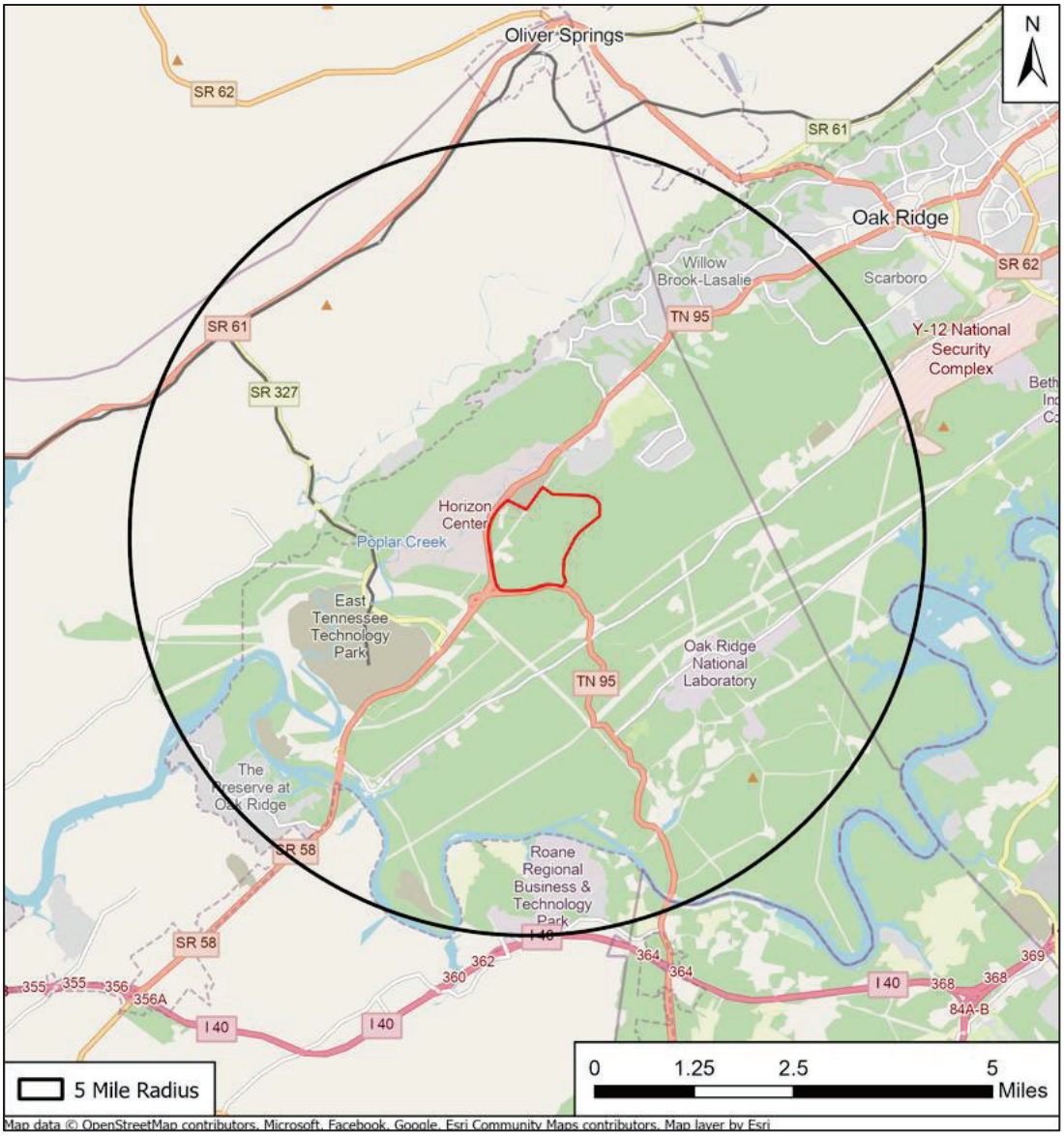

Figure 1-2 Aerial View of the Facility Site Region within 5mi

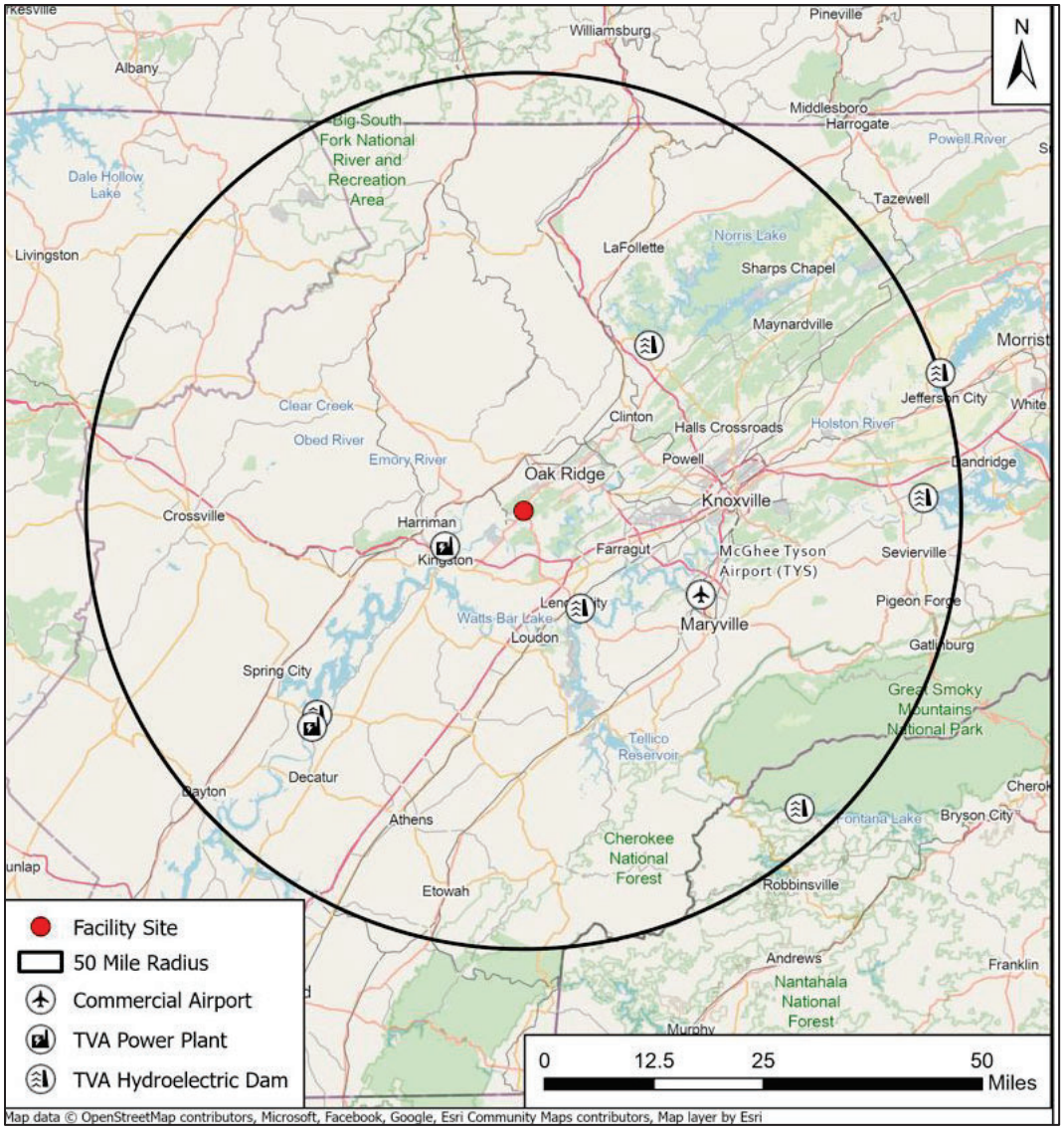

Figure 1-3 Major Population Centers near the Facility within 50mi

Site features are well suited for the location of a uranium enrichment facility as evidenced by its favorable conditions of hydrology, geology, seismology, and meteorology as well as good transportation routes for transporting feed and product by truck.

The Facility is located in the western portion of the Site on about 100 acres of undeveloped area.

The Site lies along the east side of Tennessee State Highway 95, the Oak Ridge Turnpike. A newly constructed main access road provides access to the Facility. The Site is comprised mostly of relatively flat and gently sloping forested surfaces with a dominant ridge extending from the northeastern part of the site to the northeast. Elevations at the Site range from 2,500 ft to 775 ft above mean sea level (AMSL). The overall slope direction is to the southwest.

The nearest community is the city of Oak Ridge, approximately 6.7 miles northeast of the Site, with a population of 31,402 (USCB 2020). There are no residences, schools, stores or other population centers within a 1 mile radius of the site.

Additional details of proximity to nearby populations are provided in Section 3.1 of the ER.

1.1.4 Descriptive Summary of Licensed Material

The Facility handles SNM of ^{235}U contained in uranium enriched above natural but less than or equal to 10.0% in the ^{235}U isotope. The ^{235}U is in the form of UF_6.

Total quantities of licensed materials to be held at the Facility is described in Table 1-1. A breakdown of maximum SNM by major building and process area is listed in Table 1-2, with additional information in Ch. 6.

Table 1-1 Type, Quantity and Form of Licensed Material

Source and/or Special Nuclear Material	Chemical and/or Physical Form	Maximum amount that licensee may possess at any one time
Uranium (natural and depleted) and daughter products	Physical: Solid, Liquid and Gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	750,000,000 kg
Uranium enriched in isotope ^{235}U up to 10% by weight and Uranium daughter products	Physical: Solid, Liquid, and Gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	[ ]
^{99}Tc, transuranic isotopes and other contamination	Any	Amount that exists as contamination as a consequence of the historical feed of recycled Uranium at other facilities[1]

Notes:

1. To minimize potential sources of contamination of UF_6, such as ^{99}Tc, OE will require UF_6 suppliers to provide Commercial Natural UF_6 in accordance with ASTM C787 (ASTM 2025). In addition, cylinder suppliers will be required to preclude use of cylinders that, in the past, have contained reprocessed UF_6, unless they have been decontaminated. Periodic audits of suppliers will be performed to provide assurance that these requirements are satisfied.

Table 1-2 Maximum SNM per Major Building and Process Area

1.1.4.1 Waste Discharge Points

The principal waste types generated at the Facility are:

- Solid radioactive waste includes mainly filters and media used in the chemical traps, and personal protective equipment used in controlled areas. Uranium contamination is typically in the form of uranyl fluoride or other uranium compounds resulting from UF_6 hydrolysis.
- Solid non-radioactive waste includes general industrial waste and non-contaminated maintenance waste generated outside controlled areas.
- Liquid radioactive waste coming from decontamination solutions used for cleaning UF_6 handling systems. These liquids may contain dissolved uranium compounds and are collected, characterized, and transferred for appropriate treatment or disposal.
- Liquid non-radioactive waste, including sanitary wastewater and non-contact cooling water. These streams are segregated from process systems and do not contain licensed material.
- Gaseous waste, exhausted from ventilation of controlled areas and process off gas streams. These gases may contain trace quantities of uranium particulates or HF under abnormal conditions but are routed through engineered controls such as cold traps and chemical traps, scrubbers, and HEPA filtration systems.

Based on facility design and operational controls, no environmental discharge points exist for licensed radioactive material or hazardous UF_6 related chemicals during normal operations. All potential discharge pathways are engineered for containment, monitoring, and mitigation.

UF_6 process systems are fully enclosed and maintained under negative pressure with continuous monitoring. UF_6 or HF generated through incidental UF_6 hydrolysis is captured by cold traps, chemical traps, scrubbers, or filtration systems.

Ventilation exhaust is released through monitored stacks to verify that uranium concentrations remain below detection limits during normal operations. These exhaust points are not considered radioactive discharge points under normal operating conditions.

Sanitary wastewater and non-contact water is discharged to a permitted municipal treatment facility under applicable state permit.

A full description of the waste impacts is discussed in Section 4.13 of the Environmental Report (OE 2026).

1.2 INSTITUTIONAL INFORMATION

This section provides the applicant's corporate identity and location, applicant's ownership organization and financial information. Also, the type, quantity, and form of licensed material to be used at the facility, and the type(s) of license(s) being applied for are discussed.

1.2.1 Corporate Identity and Ownership

1.2.1.1 Applicant

The Applicant's name, address, and principal office are as follows:

Orano Enrichment USA LLC
4747 Bethesda Avenue Suite 1001
Bethesda, Maryland 20814

1.2.1.2 Organization and Management of Applicant

Orano (SA) is a corporation founded under the laws of France and governed by an Executive Board. Orano SA offers products and services covering the whole nuclear fuel cycle including uranium mining, chemistry and enrichment, as well as waste management, recycling, decommissioning, radiological protection, maintenance, logistics services, and medical nuclear isotopes.

Since its latest capital increase on October 24, 2024, ownership of Orano SA's share capital is as follows:

- French State: 90.33%
- Mitsubishi Heavy Industries: 4.83%
- Japan Nuclear Fuel Limited: 4.83%

Orano Chimie Enrichissement (SAS) is a 100% subsidiary of Orano (SA) and is headquartered in Châtillon, France. Orano Chimie Enrichissement (SAS) is responsible for uranium chemistry and enrichment activities worldwide.

Orano Chimie-Enrichissement USA Holding (SAS) is a 100% subsidiary of Orano Chimie Enrichissement (SAS) and is headquartered in Châtillon, France.

Orano Enrichment USA Holding LLC is a 100% subsidiary of Orano Chimie-Enrichissement USA Holding (SAS) and is headquartered in the US.

Orano Enrichment USA LLC (OE) is a 100% subsidiary of Orano Enrichment USA Holding LLC and is headquartered in Bethesda, MD, USA. OE will be operating the Facility once licensed and constructed.

The President and Chief Executive Officer of Orano Enrichment USA LLC is a naturalized citizen of the United States of America and a citizen of France.

NRC notified OE's Facility Security Officer by a letter dated January 13, 2026 that Foreign Ownership, Control and Influence (FOCI) requirements had been waived by NRC after a FOCI review, in conformity with previous decisions regarding users of the ETC technology: March 20, 2012 for AREVA Enrichment Services, LLC.

1.2.1.3 Legal Site Description

SSP-2 was transferred from the U.S. DOE to the Industrial Development Board of the City of Oak Ridge. There is a portion of this parcel that is designed as the White Wing Scrapyard which will still fall under U.S. DOE responsibility as noted in DOE/OR/01-2970 & D1 (DOE 2024b).

The legal description is subject to the final deed with the Industrial Development Board of Oak Ridge and contains the Tract 1A and 1B of SSP-2 (DOE 2025a).

1.2.2 Financial Qualifications

OE estimates the total cost of the Facility to be approximately $5 billion (in 2025 dollars), excluding escalation, contingency, interest, tails disposition, decommissioning, and any replacement equipment required during the life of the Facility.

Pursuant to 10 CFR 70.23(a)(5), OE is required to demonstrate that it is financially qualified to carry out the activities proposed in its application. That approach is as follows:

- Construction of each incremental phase of the Facility shall not commence before funding for that increment is available or committed. Of this funding, prior to constructing such increments, OE must have in place commitments for one or more of the following:
  - Equity contributions from OE or its parents,
  - Commitment from the parent company to provide the necessary funds for the project,
  - Lending arrangements that solely or cumulatively are sufficient to ensure funding for the particular increment's construction costs.

OE shall make available for NRC inspection documentation of both the budgeted costs for each incremental phase and the source of funds available or committed to pay those costs.

- Operation of the Facility shall not commence until OE 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).

OE shall in accordance with 10 CFR 140.13b, prior to and throughout operation, have and maintain nuclear liability insurance in the type and amount to cover 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. Orano has initiated conceptual discussion with ANI but does not have any indication to provide regarding the expected coverage or insurance details at this time

The effective date of this insurance will be no later than 45 days prior to the date that OE takes possession of licensed nuclear material.

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, 10 CFR 30.25 and 10 CFR 40.36 is described in detail in Chapter 10, Decommissioning.

1.2.3 Characteristics of the Material

OE proposes to acquire, deliver, receive, possess, produce, use, transfer, or store byproduct material as described in 10 CFR 30.4, source material as described in 10 CFR 40.4, and special nuclear material (SNM) meeting the criteria of low strategic significance as described in 10 CFR 70.4 are provided in Table 1-1.

Source material will be natural or depleted uranium in the chemical form of UF_6. To minimize potential sources of contamination of UF_6, such as 99Tc, OE will require UF_6 suppliers to provide Commercial Natural UF_6 in accordance with ASTM C787 (ASTM 2025). Byproduct material will consist of discrete sources used for instrument calibration. OE will submit a request to amend the Materials License to add the proposed quantities and types of sealed and unsealed instrument calibration sources to the possession limits. Subsequently, the License Application maybe revised to incorporate additional byproduct material.

OE does not propose possession of any reflectors or moderators with special characteristics.

1.2.4 Authorized Uses

This license application is for a material license issued under 10 CFR Parts 30 , 40, and 70, to construct, own, use and operate an uranium enrichment Facility with enrichment level up to 10.0 $^{w}/_{o}$ ^{235}U.

1.2.5 Special Exemptions or Special Authorizations

In accordance with 10 CFR 40.14 and 10 CFR 70.17, OE requests exemptions from certain provisions of 10 CFR 40.36 paragraph (d), and 10 CFR 70.25 paragraph (e). Specifically, 10 CFR 40.36(d) and 10 CFR 70.25(e) both state 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*...."

As stated in Chapter 10 of the License Application, since OE intends to sequentially install and operate modules of the enrichment equipment over time providing financial assurance for decommissioning during the operating life of the Facility at a rate that is in proportion to the decommissioning liability for these facilities as they are phased in, this satisfies the requirements of this regulation without imposing the financial burden of maintaining the entire financial coverage for facilities and material that are not yet in existence. The same basis applies to decommissioning funding assurance for DUF_6. As also stated in Chapter 10 of the License Application, OE proposes to provide financial assurance for the disposition of DUF_6 at a rate in proportion to the amount of accumulated DUF_6 onsite up to the maximum amount of the DUF_6 produced by the Facility.

The justification for this proposal to provide decommissioning funding assurance on a forward looking incremental basis is OE's commitment to update the decommissioning cost estimates and to provide to the NRC a revised funding instrument for facility decommissioning at a minimum prior to the operation of each facility module. With respect to the DUF_6, OE commits to updating the decommissioning cost estimates on an annual forward-looking incremental basis and to providing the NRC revised funding instruments that reflect these projections of DUF_6 production.

The long-term nature of enrichment contracts allows OE to accurately predict the production of DUF_6. If any adjustments to the funding assurance were determined to be needed during the annual period due to production variations, they would be made promptly and a revised funding instrument would be provided to the NRC.

OE requests that exemptions from the provisions of 10 CFR 40.36(d) and 10 CFR 70.25(e) described above be granted.

In support of this request, OE provides the following information relative to the criteria in 10 CFR 40.14 and 10 CFR 70.17. This exemption is justified for the following reasons:

- It is authorized by law because there is no statutory prohibition on incremental funding of decommissioning costs.
- The requested exemption will not endanger life or property or the common defense and security for the following reasons: the modular aspects of the Facility allow enrichment operations to begin well before the full capacity of the Facility is reached. Thus, the decommissioning liability is incurred incrementally as more centrifuge machines are added to the process, until full capacity of the facility is reached; at which point the DUF_6 are generated at a relatively constant rate throughout the life of the Facility. As such, requiring full funding for decommissioning liability, to include DUF_6 disposition, incurred over the lifetime of the

Facility, at the time of initial license issuance, produces an unnecessary financial burden on the licensee.

- Furthermore, incremental funding of decommissioning costs, to include DUF_6 disposition, is justified based upon OE's commitments to update the cost estimates and provide a revised funding instrument for decommissioning and DUF6 disposition prior to operation of each additional increment of capacity on process gas, and after full capacity has been reached to annually adjust the cost estimate for DUF_6 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 DUF_6 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 other Uranium Enrichment Facilities in the United States.
- 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.

1.2.6 Protection of Safeguard Information

This license application includes Safeguards Information (SGI) that is submitted separately. Specifically, the Physical Security Plan required as part of Chapter 13 is SGI. SGI is protected in accordance with 10 CFR 73.21-23. The Facility will handle SGI, specifically, the Physical Security Plan and the Safeguards Contingency Plan for Transport of Special Nuclear Material for transportation of radioactive materials.

1.2.7 Protection of Classified Information

This license application includes classified information that is submitted separately. Classified information is protected in accordance with 10 CFR 95 and 32 CFR 117. The Facility will handle classified matter, specifically, Uranium centrifuge technology in the form of SECRET-Restricted Data (RD) and CONFIDENTIAL-RD. OE personnel may have access to NSI related to facility threats; however, they will not generate, store, or process any NSI. Facility threat assessments may be provided by the NRC or the FBI.

1.2.8 Period of Time for Which the License Is Requested

OE is requesting a License for 40 years per NRC policy and practice SRM-SECY-06-0186 [ML062700110] (NRC/SECY 2006).

1.3 SITE DESCRIPTION

The site OE selected for the new facility is in eastern Tennessee, located in Roane County on the eastern side of Tennessee State Highway 95, and about 6.7 miles southwest of the city of Oak Ridge. The Site is currently situated on land owned by the Industrial Development Board of the City of Oak Ridge and is specifically identified as the Oak Ridge SSP-2. SSP-2 is approximately 2.9 mi southwest of the DOE's Oak Ridge National Laboratory (ORNL). The surrounding area is part of the DOEs ORR and can be characterized as industrial, with a large portion of the area being greenfield sites.

The following sections provide a high level summary of the extensive site description detail found in Chapter 3 of the ER (OE 2026).

1.3.1 Site Geography

Site features are well suited for the location of a uranium enrichment facility as evidenced by the favorable conditions of hydrology, geology, seismology and meteorology as well as good transportation routes for transporting feed, product, and tails by truck.

1.3.1.1 Site Location Specifics

The approximate center of the Facility is located at latitude 35 degrees, 57 minutes, 4 seconds North and longitude 84 degrees, 21 minutes, 26 seconds West. The Facility will occupy about 100 acres of the western side of the Site.

More information on the site location can be found in the ER.

1.3.1.2 Major Nearby Highways

The Site is located east of Tennessee State Route 95 (SR-95), which is a multi-lane State highway. West of the Facility at an interchange, SR-95 turns south toward Lenior City and connects to Interstate 40, continuing south the route turns into Highway 321 which connects to Interstate 75. From the SR-95 interchange, continuing southwest, it connects to State Route 58. SR 58 is a north-south state highway that serves as a major route for many communities in Roane, Meigs, and Hamilton counties. State Route 58 joins Interstate 40 for part of its route in Roane County, from Kingston (exit 352) east to Oak Ridge (exit 356) west of Oak Ridge. East of the Facility SR-95 turns into Highway 61 and connects to Interstate 75 after passing through the cities of Oak Ridge and Clinton, Tennessee. Interstates 40 and 75 interchange in two locations: in Knoxville, Tennessee, for Interstate 75 northbound into Kentucky, and north of Lenoir City for Interstate 75 southbound toward Chattanooga, approximately 8 mi southeast of the Facility.

The primary access to the site would be from SR-95 at Novus Drive north of the SR-95 intersection with SR 58.

For more detailed information, refer to the Section 3.2 of the ER.

1.3.1.3 Nearby Bodies of Water

The Facility site is located within the Valley and Ridge physiographic province. The Facility is located within the East Fork Valley, a valley between two parallel ridges, Blackoak Ridge on the north and East Fork Ridge to the south. East Fork Poplar Creek flows southwesterly through this valley and just west of the Facility site and just east of the TRISO-X facility. The Facility and the East Fork Poplar Creek (EFPC) channel are separated by Highway 95. The EFPC watershed extends to the northeast and encompasses much of the western portion of the City of Oak Ridge.

EFPC is a tributary to Poplar Creek, which flows southwesterly in the valley north of East Fork Valley and Blackoak Ridge. The EFPC confluence with Poplar Creek, a tributary to the Clinch River, is located southwest of Facility. Numerous small tributaries drain to EFPC from the ridges along the East Fork Valley. Bear Creek is downstream of the Facility and flows southwesterly in the valley south of the East Fork Valley and cuts through Pine Ridge to flow into EFPC.

For more detailed information, refer to the Section 3.4 of the ER.

1.3.2 Demographics

The city of Oak Ridge is the nearest large population center and is located approximately 6 mi northeast of the site. Rockwood Tennessee has the closest commercial airport, McGhee Tyson, at approximately 19 miles from the site.

Demographics are further discussed in the Section 3.10 of the ER.

1.3.2.1 Census Data

The Facility is located within the City of Oak Ridge in Roane County, Tennessee. According to the 2020 Census the population of Oak Ridge was 31,402 and the population of Roane County was 53,404. The populations for the surrounding counties within the Region of Influence (ROI) are as follows: Knox County 478,971; Anderson County 77,123; Loudon County 54,886; and Morgan County 21,305. The cumulative population for the five counties within the ROI accounts for 9.92% of the population of Tennessee.

This is further discussed in the Section 3.10 of the ER.

1.3.2.2 Nearby Populated Areas

The ROI consists of Roane, Morgan, Anderson, Knox, and Loudon counties. The site is located in the City of Oak Ridge in Roane County, Tennessee. Other population centers are located the approximate driving distance from the site:

- Lenoir City, Loudon County: 8.3 miles south
- Kingston, Roane County: 12.2 miles southwest
- Harriman, Roane County: 17.5 miles west
- Farragut, Knox County: 18.9 miles southeast
- Knoxville, Knox County: 32.8 miles east

This is further discussed in the Section 3.10 of the ER.

1.3.2.3 Proximity to Nearby Public Facilities

There are no nearby (within 1 mile) public facilities that need to be considered. Further information is discussed in the Section 3.1 of the ER.

1.3.2.4 Nearby Industrial Facilities (Includes Nuclear Facilities)

Nuclear Facilities

NAC Philotechnics is located approximately 0.9 mi northwest of the Facility and X-Energy's TRISO-X facility will be located approximately 1.0 mi northwest of the Facility.

Non-Nuclear Facilities

The city of Oak Ridge is located approximately 6.4 mi northeast of the site. The nearest residence is 1.2 mi northeast of the Facility. Public use areas include a hiking trail north and west of Facility.

The nearest offsite agricultural farm is approximately 3.9 mi to the south-southwest of the Facility. The nearest feedlot and dairy operations are about 18.4 mi south-southwest of the Facility.

The Oak Ridge Enhanced Technology and Training Center and the Enhanced Emergency Response Facility are located approximately 0.8 mi north of the Facility. The ORNL Carbon Fiber Technology Facility is located approximately 0.5 mi west of the Facility.

Refer to ER Chapter 2 for additional information.

1.3.2.5 Land Use Within Five Mile Radius, Uses of Nearby Bodies of Water

The dominant land uses in the region are forest lands, developed lands agriculture, and waters or wetlands. Refer to ER Section 3.1 for additional information.

The Facility will obtain water for both potable and industrial use from the City of Oak Ridge public water distribution system. All wastewater is discharged to the City of Oak Ridge municipal sanitary sewer collection system.

Refer to ER Section 3.4 for additional information.

1.3.3 Meteorology

The meteorological conditions at the Facility have been evaluated and summarized in order to characterize the site climatology and to provide a basis for predicting the dispersion of gaseous effluents.

In support of the Facility, meteorological towers with instrumentation set at various heights are located on the reservation of the ORNL and give a detailed estimate of meteorological conditions that would exist at the site. A first set of meteorological tower data was obtained from towers surrounding the Site and a second set was obtained from nearby airports. By obtaining data from each of these towers, a detailed pattern for the climate is obtained and this data is then used for release / dispersion calculations.

Refer to ER Section 3.6 and ISAS Section 1.1.5 for additional information.

1.3.3.1 Primary Wind Direction and Average Wind Speeds

The eleven meteorological towers located on the ORNL property were used to gather wind data for the Site. Wind data was obtained from the ORR Meteorology web page for these towers (ORNL n.d.). Table 1-3 lists data from the highest and lowest Met towers AMSL used to gather wind data. All other towers lie at elevations between these towers, so a correlation can be drawn with regards to the wind speed. The trend for the monthly/annual wind speed for all towers is thus very consistent, as shown in Figure 3-45 of the OE ER. Data is compiled from Table 3-9 and Table 3-13 of the ER.

Table 1-3 ORNL Met Tower Wind Data

Tower	Elevation (AMSL)	Instrument Level (m AGL)	Distance to Site (mi)	Period of Record	Max Wind Gust	Annual Average (mph)	Primary Wind Direction
Tower F	~1147 ft	10	3.03	2017-2021	66.7	4.58	WSW, NE, ENE, W
Tower K	~795 ft	60	2.49	2014-2016	79.2	5.78	SW, ENE, WSW
Tower M	~770 ft	10/15	3.36	2014-2020	62.3	1.98	NNE, SW, N, NE
Tower S	~1153 ft	10/15	6.11	2014-2021	Not measured	6.57	NE, SW, WSW
Tower Y	~954 ft	15	6.32	2014-2021	Not measured	4.23	ENE, SW, WSW, NE

These data are discussed and analyzed in Section 3.6 of the ER.

1.3.3.2 Annual Precipitation – Amounts and Forms

The type of precipitation that occurs at the Site varies with the seasons. Convective showers and thunder showers occur in the summer. Precipitation during the spring and fall can be characterized as showery or as steadier rainfall. Winter precipitation is typically in the form of snow.

Precipitation averages about 55.8 inches annually based on 30 years of data (Oak Ridge 30 Year Climate Average) (ORNL n.d.). Snowfall in the Oak Ridge area, though normally light, usually occurs from November through March with an average of about 5.70 inches of snow annually based on 30 years of data (Oak Ridge 30 Year Climate Average) (ORNL n.d.). Severe storms are relatively infrequent as the region is east of maximum tornado activity, south of the most significant snowstorms, and inland from hurricane and tropical storm tracks.

Heavy rains and significant snow melt from the nearby ridge could result in localized flooding of streams coming off the mountain. These streams have been identified, and appropriate terracing of the site and redirection of streams is planned.

Over the 74-year period of record, Roane County has had six ice storms recorded: one in 1998, one in 2005, one in 2010 and three in 2015 that occurred within one month (Feb to Mar) (NOAA 2025).

Refer to ER Section 3.6 for additional information.

1.3.3.3 Severe Weather

<u>Tornadoes</u>

The annual frequency of tornadoes in Roane County Tennessee and the surrounding nine-county area – encompassing Anderson, Cumberland, Knox, Loudon, McMinn, Meigs, Morgan, Monroe, and Rhea – is limited. The annual frequency of occurrence of a tornado in the region is 1.47 (109 tornadoes/74-years). The probability of a tornado developing at the Facility site is very small, due to the mountainous terrain that starts west and south-west of the Facility site.

The largest tornado that occurred in the aforementioned counties was an EF-4 that occurred in Meigs/McMinn on May 2, 1953. Four deaths and eight injuries were reported in Meigs County. Roane County has had three EF-0 and two EF-1 tornadoes in Oct 1977, Jun 2011, Jun 2014, Feb 1993, and Aug 2023, respectively. No reported injuries or fatalities occurred from these tornadoes (NOAA 2025).

Refer to ER Section 3.6 for additional information.

Hurricanes

Hurricanes, or tropical cyclones, are low-pressure weather systems that develop over the tropical oceans. Hurricanes are fueled by the relatively warm tropical ocean water and lose their intensity quickly once they make landfall. The Facility is not in the vicinity of any ocean and is protected by mountains; therefore hurricanes are not considered a credible threat.

Thunderstorms and Lightning Strikes

Thunderstorm are most prevalent during the months of March through August, peaking during the month of July for the site.

Numerous thunderstorm days were identified in the vicinity of the site queried from the NCDC Storm Event Database (NOAA 2025). The output from the query contained 2,484 thunderstorm days during the 74-year period of record or 33.6 thunderstorm days per year. Several individual thunderstorms may occur during each of the thunderstorm days.

Over the 74-year period of record, Roane County has had two major lightning storms occur with thunderstorms: one in 1996, and another in 2018. The 1996 lightning strike destroyed one house by initiating a house fire. The second storm occurred on July 7, 2018, with an event at 1500 local time and again at 1600 local time (NOAA 2025).

Refer to ER Section 3.6 for additional information.

1.3.4 Hydrology

The Facility site is located within the Valley and Ridge physiographic province. The Facility is located within the East Fork Valley, a valley between two parallel ridges, Blackoak Ridge on the north and East Fork Ridge to the south. The EFPC flows southwesterly through this valley and just west of the Facility site, and is a tributary to Poplar Creek, which flows southwesterly in the valley north of East Fork Valley and Blackoak Ridge. The EFPC watershed extends to the northeast and encompasses much of the western portion of the City of Oak Ridge. The EFPC headwaters are located west of the Facility and in the City of Oak Ridge.

The Facility is located adjacent to East Fork Ridge to the northeast. Topographically, the East Fork Ridge oriented northeast to southwest rises from approximately 770 feet above mean sea level (AMSL) to over 1,100 feet AMSL immediately northeast of the Facility. A smaller unnamed ridge parallel to the East Fork Ridge provides a boundary between the Facility site and the Whitewing Scrap Yard which separates the two main watersheds in the area – the East Fork Poplar Creek Watershed and the Bear Creek Watershed. Surface water discharge within the Facility falls within the East Fork Poplar Creek Watershed with all drainage to the East Fork Poplar Creek. East Fork Poplar Creek flows southwesterly through a valley along the eastern side of the Facility. This watershed divide separates surface drainage the Facility site from that of the Whitewing Scrap Yard, which limits the potential for any potential contamination from the Whitewing Scrap Yard entering the Facility site.

Refer to ER Section 3.4 for additional information.

1.3.4.1 Characteristics of Nearby Rivers, Streams, and Other Bodies of Water

A jurisdictional waters assessment was conducted at the Site.

Twenty-eight streams/relatively permanent waterways and 42 wet weather conveyances/non-relatively permanent waterways were identified at the site. Except for Bear Creek, which is a perennial stream that traverses the southwestern corner of the site, the onsite streams are Fork Poplar Creek

The wet weather conveyances/non-relatively permanent waterways are typically headwater ephemeral channels located within the forested valleys through the site.

Seventeen wetlands were identified on the site. All but three wetlands are palustrine forested features, with the remaining three being palustrine emergent wetlands.

Refer to ER Section 3.4 for additional information.

1.3.4.2 Depth to Groundwater Table

Groundwater monitoring wells have not been installed at the Site. Observations during geotechnical drilling indicated that groundwater was absent during the time of drilling (Kiewit Engineering Group, Inc. 2025). Geotechnical borings completed at the Facility site ranged in depth from approximately 15 feet to 90 feet below ground surface during three phases of drilling. Groundwater was not encountered in any of the Phase 2 and Phase 3 borings. The elevation of the bottom of the borings ranged from approximately 735 to 793 feet AMSL, indicating that groundwater, if present, is well below the planned building grade of 825 feet AMSL.

The lack of onsite monitoring wells precludes the use of dye tracing and calculation of groundwater velocities.

The lack of groundwater encountered during the geotechnical investigation indicates that groundwater was not present at the site in the shallow subsurface during drilling (Kiewit Engineering Group, Inc. 2025). It is anticipated that water during construction is perched water associated with rainfall events. The stable groundwater table will not be encountered during grading. Excavations into bedrock along the northeast portion of the Facility site may encounter water especially after rainfall events, given the amount of bedrock proposed to be removed in this area. Dewatering using sumps and pump, and ditches is the most productive method to manage construction water.

Refer to ER Section 3.4 for additional information.

1.3.4.3 Groundwater Hydrology

Two broad hydrologic units have been identified on the ORR: (1) the Knox Aquifer, which includes the Maynardville Limestone and is highly permeable, and (2) the ORR aquitards, which consist of less permeable geologic units.

Groundwater flow in the area of the Facility site is controlled by the East Fork Ridge and the associated surface water drainages, and an unnamed ridge located between the Facility site and the White Wing Scrap Yard.

Flow paths in the active-flow zones (particularly in the aquitards) are relatively short, and nearly all groundwater discharges to local surface water drainages on the ORR.

Refer to ER Section 3.4 for additional information.

1.3.4.4 Design Basis Flood Events Used for Accident Analysis

The Facility site lies on the Site, at the base of the White Oak Mountain which is one of a series of paralleling ridges running approximately north-northeast in the Tennessee Valley between the Cumberland Plateau/Mountains to the west and the Blue Ridge and Smoky Mountains to the east. The ridge averages 1,368 feet AMSL, the highest point being 1,495 feet AMSL. The Facility site is approximately 800 to 825 feet AMSL. The Clinch River lies approximately 3.3 miles south-west of the Facility site, which is approximately 750 feet AMSL. This location, on the Clinch River, is approximately 24 miles southwest of the Norris Dam and Norris Reservoir. While a breach of the dam (retaining $1.11x10^{11}$ ft^3 of water) would not cause a flooding problem for the Facility site due to the elevation difference between the site and the Clinch River, heavy rains and significant snow melt from the nearby ridge could result in localized flooding of streams coming off the mountain. These streams have been identified, and appropriate terracing of the site and redirection of streams is planned.

The design basis flood events used for accident analysis is described in the ER Section 3.4. The Facility and the area surrounding the site are defined as “areas determined to be outside of the 0.2 percent annual chance floodplain”, also known as the 500-year floodplain, as defined by the National Flood Insurance Program (Figure 3-26 of the ER). In addition, FEMA floodplain hazard classifies the Site as an Area of Minimal Flood Hazard (Figure 3-28 of the ER) (FEMA 2007).

A floodplain study for East Fork Poplar Creek (EFPC) was completed by the Tennessee Valley Authority (TVA) in 1991. The TVA floodplain study demonstrates that the maximum 100-year flood elevation along EFPC in the vicinity of the Facility is 764.9 ft. mean sea level at Mile 2.94 upstream of the Facility, and 759.9 ft. at Mile 2.14 (Figure 3-27 of the ER). The EFPC did not show the 100-year and 500-year floodplain extending onto the Facility.

Refer to ER Section 3.4.2 and 3.6 for additional information.

1.3.5 Geology

The Site is located in the Valley and Ridge Physiographic Province, which is characterized by a series of parallel narrow, elongated ridges and valleys that follow a northeast-to-southwest trend The Valley and Ridge Physiographic Province has developed on thick, folded beds of sedimentary rock deposited during the Paleozoic era. The long axes of the folded beds control the shapes and orientations of a series of long, narrow parallel ridges and intervening valleys.

The geology of the study area is complex as a result of extensive thrust faults and folds. The Facility is underlain by bedrock of the Chickamauga Group, which is primarily a limestone with layers of siltstone. Northeast to the Site are rocks of the Rockwood Formation (southwest of the ORETTC site) Clastic bedrock of the older Rome Formation has been placed over the calcareous rocks of the Chickamauga Group and the younger clastic rocks of the Rockwood Formation by the White Oak Mountain thrust fault, which trends generally southwest to northeast in the vicinity of SR 58.

Refer to ER Section 3.3 for additional information.

1.3.5.1 Characteristics of Soil Types and Bedrock

The Natural Resources Conservation Service (NRCS) soil survey database identified 34 soil map units digitally within the 1.9-mi radius of the Facility. Of these, only the Armuchee silt loam and the Townley silt loam are present in the immediate vicinity of the Facility assessment area (USDA 2025a) (USDA 2025b).

The NRCS soil survey database depicts nine soil types within the Facility (USDA 2025a) (USDA 2025b). Five are Ridge/Upland Soils and four are Terrace/Floodplain Soils. In general, soils at the site are well to moderately well drained and non-hydric, except for the Capshaw silt loam, Colbert-Lyerly-Rock outcrop, and Hamblen silt loam which are well drained and predominately non-hydric having hydric components less than six percent (CEC 2025).

Refer to ER Section 3.3.3 for additional information.

1.3.5.2 Earthquake Magnitudes and Return Periods

Although major thrust faults are numerous in the vicinity of the study area, these faults are associated with mountain building episodes that ended more than 200 million years ago. These faults are no longer active, but stress stored up at depth in these rocks is periodically released as minor earthquakes. Since 1973, 139 earthquakes have been recorded within 62 miles of the Site with the highest magnitude of 4.7 occurring on November 30, 1973, 0.62 miles northeast of Alcoa, Tennessee, which is 24.6 miles southeast from the Facility.

The USGS rates ground motions using peak ground acceleration, which is the maximum acceleration experienced during the course of an earthquake and is measured in units of acceleration due to gravity (g). The seismic map for 2018 indicates that the study area is located in an area with a moderate seismic hazard class rating: 0.34 g peak horizontal ground acceleration with a 2 percent probability of exceedance in 50 years; and 0.10 g peak horizontal ground acceleration with a 10 percent probability of exceedance in 50 years (USGS 2025b).

Refer to ER Section 3.3 and ISAS Section 1.1 for additional information.

1.3.5.3 Other Geologic Hazards

Landslide hazard

The USGS has identified zones of varying landslide susceptibility within the conterminous United States. Erosion has produced steep slopes including the development of canyons throughout the site region. Persistent rainfall followed by more intense precipitation has resulted in damaging debris slides and avalanches. Many landslides have occurred on soils derived from weathered Pennsylvanian and Permian sedimentary rocks. Common forms of mass wasting in the site region consist of rockslides originating from detached rock slabs and translational landslides involving soils containing elevated groundwater under a hydrostatic head. Numerous slow-moving debris slides occur in the Valley and Ridge and Blue Ridge provinces. The Interior Low Plateaus and the Cumberland Plateau are underlain by relatively flat-lying Devonian and Mississippian shales, sandstones and limestones. The shale becomes susceptible to landsliding when weathered into clayey soils.

Slope stability analyses were completed during the geotechnical investigation for approximately 50 feet tall, 3H:1V fill slope proposed on the south side of the project where the existing valley will be filled (Kiewit 2025a). Based on the established correlation between friction angle and soil plasticity index, the friction angle for the onsite clay is estimated to range between 25 and 30 degrees. To maintain a conservative approach in the slope stability analysis, the lower bound values of the friction angle were adopted. Groundwater was assumed at the soil and bedrock interface. Based on these analyses, the proposed embankment outslope of 3H:1V has a calculated factor of safety greater than 1.5. No other slope stability concerns were identified based on existing site conditions and proposed site grading. Based on the geotechnical analyses, multiple recommendations were made to OE to incorporate into the ongoing design, and are listed in Section 3.3.7 of the ER.

Refer to ER Section 3.3 for additional information.

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Chapter 2 – Organization & Administration

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2.0 ORGANIZATION AND ADMINISTRATION

This chapter describes the management system and administrative procedures for the effective implementation of Health, Safety, and Environmental functions at the Facility. The chapter presents the organizations responsible for managing the design, construction, operation, and decommissioning of the facility. The key management and supervisory positions and functions are described including the personnel qualifications for each key position at the facility.

Orano Enrichment USA LLC (OE) is responsible for maintaining a safe workplace for its employees and ensuring operational compliance within the terms and conditions of the license and applicable regulations.

The Facility organization reflects Orano's experience in operating fuel cycle facilities, including over more than a decade of commercial service at the Georges Besse 2 Enrichment Facility.

Managerial positions that have principal responsibilities important to environmental, health, safety, safeguards, security, and quality for the Facility are described in this chapter.

2.1 ORGANIZATIONAL STRUCTURE

The OE organizational structure is described in the following sections. The organizational structure indicates the lines of communication and management control of activities associated with the engineering, procurement, construction, operation, and decommissioning of the facility.

2.1.1 Corporate Functions, Responsibilities, and Authorities

OE is responsible for the design, quality assurance, construction, operation, and decommissioning of the enrichment facility. The President has overall responsibility for these functions of the Facility.

The Project Director (or equivalent senior manager) and the Quality Assurance (QA) Manager report to the President. The Project Director is accountable for design, construction, and procurement and for initial startup of the Facility. The Project Director will have oversight over the roles described in the following sections.

For matters related to QA, the QA manager has a direct access to the President, as shown in Appendix A of the QAPD.

During the operating phase, OE commits to have the Plant Manager and the Safety Review Committee (SRC) also report to the President to ensure independence.

2.1.2 Engineering, Procurement and Construction Phase Organization

During the Engineering, Procurement and Construction (EPC) phase, the Project Director is responsible for managing the engineering, construction, and procurement activities.

The Project Director has designated the Environmental, Health, Safety & Licensing (EHS&L) Manager the responsibilities to ensure compliance with licensing commitments, industrial safety, radiation protection, physical security, nuclear material control and accounting (MC&A) and safeguards, nuclear safety and criticality, and emergency preparedness. The EHS&L Manager has the authority to shut down the construction, commissioning, and start-up activities during the EPC phase.

The QA Manager has the responsibilities for independent oversight and implementation of the QAPD. The QA Manager has a direct access to the President.

The lines of communication of key management positions within the engineering, procurement and construction organization are shown in the QAPD.

Position descriptions of key management personnel with EHS&L responsibility in the engineering, procurement and construction organization will be accessible to all affected personnel and the NRC.

2.1.3 Transition from EPC phase to Operations

OE is responsible for the design, quality assurance, construction, testing, initial startup, operation, and decommissioning of the facility.

As portions of the Facility gradually get constructed and commissioned, the focus of the organization will shift from construction, commissioning and start-up to operation of the Facility. OE will staff the Facility operating organization to ensure a smooth transition from construction to operational activities.

As the construction of systems is completed, the systems will undergo acceptance testing as required by procedures, followed by turnover from the construction organization to the operations organization by means of a detailed transition plan. The turnover will include the physical systems and corresponding design information and records. Following turnover, the operating organization will be responsible for system maintenance and configuration management. The design basis for the Facility is maintained during the transition from construction to operations through the configuration management system described in Chapter 11.

2.1.4 Operating Organization

Following the transition from EPC phase to Operations, OE has direct responsibility for operation and maintenance of the Facility.

The President has overall responsibility for the operation of the Facility. He is also responsible for the QA Program. In the discharge of these responsibilities, he directs the activities of the following key personnel:

- Plant Manager
- Safety Review Committee

The Plant Manager reports to the President and is responsible for the operation and maintenance of the Facility. In the discharge of these responsibilities, he directs the activities of the following key personnel:

- Operations Manager
- EHS&L Manager
- QA Manager

The responsibilities, authorities, and lines of communication of key management positions within the operating organization are discussed in Section 2.2, Key Management Positions.

The QA Manager has the authority and responsibility to contact directly the President with any Quality Assurance concerns during operation.

Position descriptions for key management personnel in the operating organization will be accessible to all affected personnel and to the NRC.

2.2 KEY EHS&L AND PLANT OPERATIONS POSITIONS

This section describes the functional positions responsible for managing the operation of the facility. The facility is staffed at sufficient levels prior to operation to allow for training, procedure development, and other pre-operational activities.

The responsibilities, authorities, and lines of communication for each key management position are provided in this section. Responsible managers have the authority to delegate tasks to other individuals; however, the responsible manager retains the ultimate responsibility and accountability for implementing the applicable requirements.

The Corporate Organization and lines of communication are shown in the QAPD.

2.2.1 Operating Organization

The functions and responsibilities of key facility management are described in the following paragraphs.

The minimum qualification requirements for the facility functions that are directly responsible for its safe operation shall be as outlined below.

The nuclear experience of each individual shall be determined to be acceptable by the President. “Responsible nuclear experience” for these positions shall include (a) responsibility for and contributions towards support of facility(ies) in the nuclear fuel cycle (e.g., design, construction, operation, and/or decommissioning), and (b) experience with chemical materials and/or processes.

The actual qualifications of the individuals assigned to the key facility positions will be maintained in the employee personnel files or other appropriate file at the facility.

A. President

The President has overall responsibility for the design, construction, startup, and operation of the Facility. They are also responsible for the QA Program and for determining the status, adequacy, and effectiveness of its implementation.

The assignment of individuals to the Manager positions reporting directly to the President, and to positions on the SRC shall be approved by the President. The President may approve different experience requirements for key positions. Approval of different requirements is done in writing and only on a case-by-case basis.

The President delegates relevant technical tasks to the roles listed below.

B. Plant Manager

The Plant Manager shall be appointed by, and report to the President. The Plant Manager has direct responsibility for operation of the facility in a safe, reliable, and efficient manner. The Plant Manager is responsible for the protection of the facility staff and the general public from radiation and chemical exposure and/or any other consequences of an accident at the facility and also bears the responsibility for compliance with the facility license. The Plant Manager or designee(s) have the authority to approve and issue procedures.

The President shall appoint the Plant Manager as the manager of the Facility. The Plant Manager shall be knowledgeable of the enrichment process, enrichment process controls and ancillary processes, criticality safety control, chemical safety, industrial safety, and radiation protection program concepts as they apply to the overall safety of a nuclear facility.

The Plant Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and ten years of responsible nuclear experience.

C. Quality Assurance Manager

The Quality Assurance Manager is appointed by and reports to the President and has overall responsibility for development, management, implementation, and independent oversight of the QA Program. The Facility managers and their staff who are responsible for performing quality-affecting work are responsible for ensuring implementation of and compliance with the QA Program. The QA Manager position is independent from other management positions at the Facility to ensure the QA Manager has direct access to the President for matters affecting quality.

The Quality Assurance Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and at least six years of responsible nuclear experience in the implementation of a QA program. The QA Manager shall have at least four years of experience in a QA organization at a nuclear facility.

D. Environmental, Health, Safety & Licensing Manager

The Environmental, Health, Safety & Licensing (EHS&L) Manager reports to the Plant Manager and has the overall responsibility for the development and implementation of programs addressing worker health and safety; environmental protection; and licensing and permitting. The EHS&L Manager is also responsible for maintaining compliance with safeguards; appropriate rules, regulations, and codes. This includes EHS&L activities associated with nuclear criticality safety, radiation protection, chemical safety, fire protection, environmental protection, emergency preparedness, industrial safety, and development and implementation of security programs for nuclear material control and accountability, physical protection of the facility, and protection of classified matter. The EHS&L Manager works with the other facility managers to ensure consistent interpretations of EHS&L requirements, performs independent reviews, and supports facility and operations change control reviews.

This position is independent from other operations management positions at the facility to ensure objective EHS&L audit, review, and control activities. The EHS&L Manager has the authority to order the shutdown of operations if they appear to be unsafe or non-compliant with applicable regulatory requirements and must consult with the Plant Manager with respect to restart of shutdown operations after the deficiency, or unsatisfactory condition, has been resolved.

The EHS&L Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and at least five years of responsible nuclear experience in EHS&L or related disciplines. The EHS&L Manager shall also have at least one year of direct experience in the administration of nuclear criticality safety evaluations and analyses.

E. Operations Manager

The Operations Manager reports to the Plant Manager and has the responsibility of directing the day-to-day operation of the Facility. Inherent in this responsibility is the assurance that the operations are conducted safely and in compliance with license conditions. This includes such activities as ensuring the correct and safe operation of Uranium Hexafluoride (UF_6) processes, proper handling of UF_6, and the identification and mitigation of any off normal operating conditions. The Operations Manager is also responsible for the plant maintenance function, which includes activities to ensure that Items Relied on for Safety (IROFS) are reliable and available when needed. In the event of the absence of the Plant Manager, the Operations Manager may assume the responsibilities and authorities of the Plant Manager.

The Operations Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

F. Nuclear Safety & Criticality Manager

The Nuclear Safety & Criticality Manager reports to the EHS&L Manager. He is responsible for the development and implementation of the nuclear criticality safety program. Key responsibilities include the performance of nuclear criticality safety analyses and evaluations of applicable operations involving special nuclear material and changes to those operations; establishing limits and controls based on those analyses and evaluations; ensuring the proper incorporation of limits and controls into applicable procedures and instructions; and monitoring plant compliance with nuclear criticality safety requirements.

The Nuclear Safety & Criticality Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience related to nuclear safety and criticality. The Nuclear Safety & Criticality Manager shall also have at least one year of direct experience in the administration of nuclear criticality safety evaluations and analyses. No credit for academic training may be taken toward fulfilling this experience requirement.

G. Industrial Safety & Radiation Protection Manager

The Industrial Safety & Radiation Protection Manager reports to the EHS&L Manager and has the responsibility for developing and implementing programs related to industrial hygiene and safety, chemical safety, fire protection, and radiation protection (including the As Low as Reasonably Achievable (ALARA) program). The radiation protection duties include the training of personnel in use of equipment, control of radiation exposure of personnel, continuous determination of the radiological status of the facility, and conducting the radiological environmental monitoring program.

During emergency conditions the Industrial Safety & Radiation Protection Manager's duties may also include:

- Providing Emergency Operations Center personnel information and recommendations concerning chemical and radiation levels at the facility
- Gathering and compiling onsite and offsite radiological and chemical monitoring data
- Making recommendations concerning actions at the facility and offsite deemed necessary for limiting exposures to facility personnel and members of the general public
- Taking prime responsibility for decontamination activities.

The Industrial Safety and Radiation Protection Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

H. Emergency Preparedness Manager

The Emergency Preparedness Manager reports to the EHS&L Manager. The Emergency Preparedness Manager is responsible for implementation and maintenance of the emergency preparedness including the responsibility for ensuring the Facility remains prepared to react and respond to any emergency situation that may arise, as well as coordination with Physical Security Manager and MC&A and Safeguards Manager to prevent unauthorized access to classified matter and prevent the theft of nuclear material during any emergency situation. This includes emergency preparedness training of facility personnel, facility support personnel, the training of, and coordination with, offsite emergency response organizations (EROs), and conducting periodic drills to ensure facility personnel and offsite response organization personnel training is maintained up to date.

The Emergency Preparedness Manager shall have as a minimum, a bachelor's degree in an engineering or scientific field, and five years of experience in the responsible management of emergency preparedness at a facility requiring capability similar to that required for the facility.

I. Licensing and Compliance Manager

The Licensing and Compliance Manager reports to the EHS&L Manager. The Licensing and Compliance Manager is responsible for regulatory oversight functions, regulatory and environmental compliance, facility change process, and commitment management. The Licensing and Compliance Manager is also responsible for ensuring abnormal events are reported to the NRC in accordance with NRC regulations.

The Licensing and Compliance Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

J. MC&A and Safeguards Manager

The MC&A and Safeguards Manager reports to the EHS&L Manager and has the responsibility for ensuring the proper implementation of the Fundamental Nuclear Material Control (FNMC) Plan. This position is separate from and independent of other departments to ensure a definite division between the safeguards group and the other departments.

The MC&A and Safeguards Manager shall have as a minimum, a bachelor's degree in an engineering or scientific field, and five years of experience in the management of a safeguards program for Special Nuclear Material, including responsibilities for material control and accounting.

K. Physical Security Manager

The Physical Security Manager reports to the EHS&L Manager.

The Physical Security Manager is responsible for the implementation of the Physical Security Plan, the Standard Practice Procedures Plan and managing the physical security services.

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

2.2.2 Safety Review Committee

The Facility maintains an SRC to assist with the safe operation of the facility. The SRC shall report to the President and shall provide technical and administrative review and audit of operations that could impact plant worker, public safety, and environmental impacts. The scope of activities reviewed and audited by the SRC shall, as a minimum, include the following:

- Radiation protection
- Nuclear criticality safety
- Hazardous chemical safety
- Industrial safety including fire protection
- Environmental protection
- ALARA policy implementation
- Changes in facility design or operations.

The SRC shall be composed of at least five members, including the Chairman. Members of the SRC may be from the OE corporate or technical staff. The five members shall include experts on

operations and all safety disciplines mentioned above. The Chairman, members and alternate members of the Safety Review Committee shall be formally appointed by the President, shall have an academic degree in an engineering or physical science field; and, in addition, shall have a minimum of five years of technical experience, of which a minimum of three years shall relate directly to one or more of the safety disciplines.

The SRC shall meet at least once per calendar quarter and shall conduct at least one facility audit per year for the above areas.

Review meetings shall be held within 30 days of any incident that is reportable to the NRC. These meetings may be combined with regular meetings. Following a reportable incident, the SRC shall review the incident's causes, the responses, and both specific and generic corrective actions to ensure resolution of the problem is implemented.

A written report of each SRC meeting and audit shall be forwarded to the President, the Plant Manager, and other appropriate Managers within 30 days and be retained in accordance with the records management system.

2.3 ADMINISTRATION

OE has established management measures to ensure safe operations of the Facility, detailed in Chapter 11. Specifically, as it relates Organization and Administration, this section summarizes how:

- The formal management measures required to ensure the availability and reliability of IROFS are established
- The activities essential for effective implementation of the EHS&L functions are documented in formally approved, written procedures, prepared in compliance with the QA Program
- The mechanism for reporting potentially unsafe conditions or activities to the EHS&L organization and facility management is documented and implemented
- Written agreements are established with offsite emergency resources such as fire, police, ambulance and rescue units, and medical services.

2.3.1 Configuration Management

Configuration management is provided for IROFS throughout facility design, construction, testing, and operation. Configuration management provides the means to establish and maintain a technical baseline for the facility based on clearly defined requirements. The responsibility for implementing the configuration management program falls under the Plant Manager.

Additional details on Configuration Management are provided in Chapter 11.

2.3.2 Maintenance

The maintenance program will be implemented for the operations phase of the Facility. Preventive maintenance activities, surveillance, and performance trending provide reasonable and continuing assurance that IROFS will be available and reliable to perform their safety functions. The responsibility for implementing the maintenance program falls under the Operations Manager.

The maintenance program is discussed in detail in Chapter 11.

2.3.3 Training and Qualifications

Formal planned training programs shall be established for Facility employees.

The training programs and maintenance of the training program records at the facility are the responsibility of the Operations Manager. Accurate records are maintained on each employee's qualifications, experience, training, and retraining.

Additional details on the Facility training program are provided in Chapter 11.

2.3.4 Procedures

Activities involving licensed materials will be conducted through the use of approved, written procedures. Applicable procedure and training requirements will be satisfied before use of the procedure. Procedures will be used to control activities in order to ensure the activities are carried out in a safe manner.

Chapter 11 details the use of procedures, including development, revision, and distribution and control.

2.3.5 Audits and Assessments

The QA Program requires periodic audits to confirm that activities affecting quality comply with the QA Program and that the QA Program is being implemented effectively. The assessment function includes audits and other independent assessments to verify performance. These assessments provide a comprehensive independent evaluation of activities, including activities delegated to others under the QA Program, and procedures. Personnel who do not have direct responsibility in the area being assessed conduct these assessments.

Additional details on audits and assessments are provided in Chapter 11.

2.3.5.1 Safety Review Committee

The SRC provides technical and administrative review of facility operations that could impact plant worker and public safety. Details on the SRC and the scope of activities reviewed by the SRC are provided in Section 2.2.3.

2.3.5.2 Quality Assurance Department

The QA Department conducts periodic audits of activities associated with the facility, in order to verify the facility's compliance with established procedures. Audits are conducted in accordance with the QA Program Description and as required by Chapter 11.

2.3.5.3 Facility Operating Organization

The facility operating organization shall provide, as part of the normal duties of supervisory personnel, timely and continuing monitoring of operating activities to assist the Plant Manager in keeping abreast of general facility conditions and to verify that the day-to-day operating activities are conducted safely and in accordance with applicable administrative controls.

These continuing monitoring activities are considered to be an integral part of the routine supervisory function and are important to the safety of the facility operation.

2.3.5.4 Audited Organizations

Audited organizations shall assure that deficiencies identified are corrected in a timely manner.

Audited organizations shall transmit a response to each audit report within the time period specified in the audit. For each identified deficiency, the response shall identify the corrective action taken or to be taken. For each identified deficiency, the response shall also address whether or not the deficiency is considered to be indicative of other problems (e.g., a specific audit finding may indicate a generic problem) and the corrective action taken or to be taken for any such problems determined.

Copies of audit reports and responses are maintained in accordance with the records management system.

2.3.6 Incident Investigations

Abnormal events that potentially threaten or lessen the effectiveness of health, safety, or environmental protection are identified and reported to the EHS&L Manager or designee through the Corrective Action Program (CAP) which is described in more detail in Chapter 11.

The EHS&L Manager or designee shall, through implementation of the CAP, maintain a record of corrective actions to be implemented as a result of off-normal investigations. These corrective actions shall include documenting lessons learned, and implementing worker training where indicated, and shall be tracked to completion by the EHS&L Manager or designee within the CAP.

All employees have the responsibility to initiate a corrective action should incidents or deficiencies be identified. The QA team will utilize the CAP to determine any root causes or corrective actions required to preclude recurrence for significant conditions adverse to quality.

Additional details on incident investigations are provided in Chapter 11 and QAPD Section 16.

2.3.7 Employee Concerns

Employees who feel that safety or quality is being compromised have the right and responsibility to initiate the “stop work” process in accordance with the applicable project or facility procedures to ensure the work environment is placed in a safe condition.

Employees also have access to various resources to ensure their safety or quality concerns are addressed, including:

- line management or other facility management (e.g., EHS&L Manager, Plant Manager, QA Manager)
- the Facility safety organization (i.e., any of the safety engineers or managers)
- NRC’s requirements under 10 CFR 19, Notices, Instructions and Reports to Workers: Inspection and Investigations
- OE CAP - a simple mechanism available for use by any person at the Facility site for reporting unusual events and potentially unsafe conditions or activities.

2.3.8 Records Management

Procedures are established which control the preparation and issuance of documents such as manuals, instructions, drawings, procedures, specifications, and supplier-supplied documents, including any changes thereto. Measures are established to ensure documents, including revisions, are adequately reviewed, approved, and released for use by authorized personnel.

Document control procedures require documents to be transmitted and received in a timely manner at appropriate locations including the location where the prescribed activity is to be performed. Controlled copies of these documents and their revisions are distributed to and used by the persons performing the activity.

Superseded documents are destroyed or are retained only when they have been properly labeled. Indexes of current documents are maintained and controlled.

The QA Program assigns responsibility for verifying QA record retention to the QA Manager. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures. QA records are not considered valid until they are authenticated and dated by authorized personnel.

Additional details on records management are provided in Chapter 11.

2.3.9 Written Agreements with Offsite Emergency Resources

The plans for coping with emergencies at the Facility are presented in detail in the Emergency Plan. The Emergency Plan includes a description of the facility emergency response organization (ERO) and interfaces with off-site EROs. Local fire, police, and ambulance services (City of Oak Ridge, Roane County) are required by Tennessee state law to respond to emergencies when 911 is called. Where letters of agreement are not in place, such as local police or fire responders, the use of the 911 system ensures a response to incidents at the facility.

More details are provided in Chapter 8 and in the Facility Emergency Plan.

Chapter 3 – Integrated Safety Analysis

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3.0 SAFETY PROGRAM

This Chapter is developed in accordance with NUREG 1520 (NRC 2015).

The three elements of the safety program defined in 10 CFR 70.62(a) are addressed below.

3.1 PROCESS SAFETY INFORMATION

3.1.1 OE compiles and maintains up-to-date documentation of process safety information. Written process-safety information is used in updating the Integrated Safety Analysis (ISA) to identify and understand the hazards associated with the operations and processes. The compilation of written process-safety information includes information pertaining to:

3.1.1.1 The material hazards in the enrichment process includes information on chemical and physical properties included within Safety Data Sheets (SDS) to meet the requirements of 29 CFR 1910.1200(g).

3.1.1.2 Process technology information includes block flow diagrams, simplified process flow diagrams, outlines of the process chemistry, threshold limits for safety-controlled parameters (e.g., temperature, pressure, flow, mass, and concentration), and evaluation of the health and safety consequences due to process deviations.

3.1.1.3 Equipment used in the process such as the materials of construction, piping and instrumentation diagrams (P&IDs); employed ventilation, design codes and standards; material and energy balances; Items Relied on for Safety (IROFS) (e.g., interlocks, detection, or suppression systems), electrical classifications, relief systems, and design basis.

3.1.1.4 The process-safety information described above and its integration into the ISA is maintained up-to-date by the Configuration Management program described in License Application (LA) §11.1.

3.1.2 OE has developed procedures and criteria for changing the ISA. This includes implementation of a facility change review mechanisms that meet the requirements of 10 CFR 70.72. NRC also provides voluntary guidance within Regulatory Guide 3.74 (NRC 2012). The development and implementation of safety basis reviews to accommodate the engineering change process are described in LA §11.4. Examples of change activities that require evaluation of the safety basis to meet 10 CFR 70.72 are provided by the following:

1. facility, design, and process changes;
2. all changes to the facility safety program, including the ISA, process safety information, and management measures; and
3. proposed activities that involve changes to procedures or new procedures not previously evaluated as part of a facility, design, or process change

3.1.3 OE uses personnel with the appropriate experience and expertise in engineering and process operations to maintain the ISA. The ISA Team consists of individuals who are knowledgeable in the ISA method(s) and the operation, hazards, and safety design criteria of the OE Facility's processes. Training and qualifications of individuals responsible for maintaining the ISA are described in LA §11.3 and Chapter 1 of the ISA Summary (ISAS).

3.2 HAZARD IDENTIFICATION

3.2.1 The hazard and operability (HAZOP) analysis method was used for identifying the hazards for the Uranium Hexafluoride (UF_6) process systems and Technical Support Building systems. The hazards identification process results in identification of physical, radiological or chemical characteristics that have the potential for causing harm to site workers, the public, or to the environment. Hazards are identified through a systematic review process that entails the use of system descriptions, piping and instrumentation diagrams, process flow diagrams, plot plans, topographic maps, utility system drawings, and specifications of major process equipment. In addition, criticality hazards identification were performed for the areas of the facility where fissile material is expected to be present. The criticality safety analyses contain information in ISAS §2.0 about the location and geometry of the fissile material and other materials in the process, for both normal and credible abnormal conditions. The ISA input information is included in the ISA documentation and is available to be verified as part of an on-site review.

The hazard identification process documents materials that are:

- Radioactive
- Fissile
- Flammable &/or Explosive
- Toxic
- Reactive

The hazard identification also identifies potentially hazardous process conditions. Most hazards were assessed individually for the potential impact on the discrete components of the process systems. However, hazards from fires (external to the process system) and external events (seismic, severe weather, etc.) were assessed on a facility wide basis.

3.3 INTEGRATED SAFETY ANALYSIS

3.3.1 In accordance with NUREG 1513 (NRC 2001), OE has conducted an ISA for each process and node, such that it identifies (i) radiological hazards, (ii) chemical hazards that could increase radiological risk, (iii) facility hazards that could increase radiological risk, (iv) potential accident sequences, (v) consequences and likelihood of each accident sequence and (vi) IROFS and credited attributes of safe-by-design components, including the assumptions and conditions under which they support compliance with the performance requirements of 10 CFR 70.61. A synopsis of the results of the ISA, including the information specified in 10 CFR 70.65(b), is provided in the Facility ISA Summary (ISAS).

3.3.2 OE has implemented programs to maintain the ISA and supporting documentation so that it is accurate and up-to-date. Changes to the ISAS are submitted to the NRC, in accordance with 10 CFR 70.72(d)(1) and (3). The ISA annual update process accounts for any changes made to co-located facilities, the Site, the Facility, or process operations. This update will also verify that initiating event frequencies and reliability values of IROFS assumed in the ISA remain valid. Any changes required to the ISA as a result of the update process will be included in a revision to the ISA. Management policies, organizational responsibilities, revision time frame, and procedures to perform and approve revisions to the ISA are outlined in Chapter 11. Evaluation of any facility changes or updates in the process safety information that may alter the parameters of a credible accident sequence within the ISA, as described in the ISAS, will require further assessment for ISA incorporation and NRC approval prior to change implementation. For any revisions to the ISA, personnel having qualifications similar to those of ISA team members who conducted the original ISA are used.

3.3.3 Personnel used to update and maintain the ISA and ISAS are trained in the ISA method(s) and are suitably qualified. Training and Qualification of personnel used to update or maintain the ISA are described in LA §11.3.

3.3.4 Proposed changes to the OE Facility, or its operations, are evaluated by the ISA method(s) described in the ISAS. New or additional IROFS and appropriate management measures are designated as required. The adequacy of existing IROFS and associated management measures are promptly evaluated to determine if they are impacted by changes to the facility and/or its processes. If a proposed change results in a new type of accident sequence, changes the consequences, or increases the likelihood of a previously analyzed accident sequence within the context of 10 CFR 70.61, the adequacy of existing IROFS and associated Management Measures will be re-evaluated, any necessary changes made, and any required notifications with or approvals from the NRC performed.

3.3.5 Unacceptable performance deficiencies associated with IROFS are addressed and are identified through updates to the ISA. Required notifications to the NRC will be done in accordance with the OE Facility's Incident Notification and Reporting protocols, in accordance with Appendix A of 10 CFR 70 and NRC Form 361A (NRC 2023).

3.3.6 Written procedures are maintained on site. LA §11.4 discusses the OE Procedures Program.

3.3.7 All IROFS are maintained so that they are available and reliable when needed to prevent and mitigate upset events having significant consequence &/or probability with a Risk Index value > -4.

3.4 MANAGEMENT MEASURES

IROFS are identified in the ISAS. Management Measures are utilized as safety management programs to support and/or maintain the IROFS so that the IROFS system components (i.e., Configured Items, or CIs) are available to perform their safety functions when needed. Management Measures disseminate from the OE Facility's Safety Management Programs are the principal mechanism to maintain reliability and availability of each IROFS . Management Measures are described in Chapter 11 of this license application.

3.5 HUMAN SYSTEM INTERFACE DESIGN

The human system interface (HSI) design process translates function and task requirements into HSI characteristics and functions. The HSI uses a structured methodology that guides designers in identifying and selecting candidate HSI approaches, defining the detailed design, and performing HSI tests and evaluations. The process and the rationale for the HSI design is documented and controlled under the design control process described in the OE Quality Assurance Program Description (QAPD).

3.5.1 Human System Interface Design Inputs

The HSI design is developed based on various design inputs. The following HFE program element design inputs will be considered in making design decisions:

- Operating experience review (OER),
- Functional requirements analysis (FRA) and function allocation (FA),
- Task analysis (TA), and
- Staffing analysis.

Additionally, the HSI design team considers applicable regulatory documents and codes as well as generic HFE standards and industry guidelines as discussed in the following subsections.

3.5.1.1 Analysis of Personnel Task Requirements

Several analyses, as indicated below, may be performed in the early stages of the design process to identify HSI design requirements.

3.5.1.1.1 Operating Experience Review

An OER determines how the strengths and weaknesses of the HSI technology concept impact the effectiveness of the operator when using the technology. The goal of the OER is to compare the analysis of current work practices, operational problems and issues in current designs, and industry experience with candidate technological approaches to system and HSI technology and specific supplier solutions.

3.5.1.1.2 Functional Requirement Analysis and Function Allocation

FRA and FA determine which operational functions are to be performed by automatic systems, by plant personnel, or by some combination of the two. The allocation is made based on the FRA after determining what is required to perform the function. FA evolves from FRA and results in allocating functions for the best overall accomplishment for that function.

The results of the FRA and FA are used to identify the personnel role in performance of functions to reveal the task requirements and identify the HSI design implications. These HSI design implications include insight into the information that is to be displayed and how that information is presented. This information is used in the HSI procedure and training design to make sure that adequate task support is available to the operators.

3.5.1.1.3 Task Analysis

TA is performed for procedure development and is iterated as the HSI design detail evolves and involves determining the requirements for plant personnel to successfully perform complex real-time control actions that stem from functions assigned to them as a result of the FA design effort. Actions performed by plant personnel to accomplish a common-purpose group of activities or functions are called tasks. TA requirements are a primary consideration in design of the HSI.

3.5.1.1.4 Staffing and Qualifications and Job Analysis

Staffing and qualification analysis considers the allocation of assigned operational activities, the impact of those activities on crew member roles and responsibilities, and the impact of changes to operational requirements for the operating crew as a whole.

The results of the evaluation of staffing, qualifications, and integrated work design may impact the HSI design in terms of how operational activities are allocated to crew members, including assignments that make operational activities more efficient or reduce workload, how teamwork is supported, personnel qualifications, and required staffing levels.

3.5.1.2 System Requirements

The HSI system requirements will be documented for use throughout the HSI design process. The design control process facilitates the translation of high level requirements to lower level requirements, design inputs to design outputs, and high level design features to lower level subsystem and component design features.

The HSI consists of the controls, alarms, and indications used by the operator for performance of the IROFS safety function. Many of the active engineering IROFS controls are designed to have an initial process threshold limit that communicates adverse conditions to the operator. A second process threshold limit will initiate automatic controls to retain the process within a safe condition.

3.5.1.3 Regulatory Requirements and Guidance

The HSI are designed to address the following regulatory requirements, as applicable:

- 10 CFR 70.62(d) requires in part that

  "…*engineered and administrative controls and control systems that are identified as items relied on for safety pursuant to §70.61(e) of this subpart are designed, implemented, and maintained, as necessary, to ensure they are available and reliable*

to perform their function when needed, to comply with the performance requirements of §70.61 of this subpart."

- 10 CFR 70.64(a) (10) requires that,

  "*The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of items relied on for safety.*"

  Given that the Facility design contains many IROFS that rely on human action, the instrumentation and control systems associated with these IROFS must be designed to adequately support operator task performance.

- NUREG-1513 (NRC 2001) identifies that for administrative controls (e.g., certain human actions),

  "…*the man-machine interface for that individual should be carefully designed*"

- NUREG-1520 (NRC 2015). Appendix E states how the HSI supports the functionality of the Facility's IROFS.

  "….to establish that human factors engineering (HFE) is applied to personnel activities identified as safety significant, consistent with the findings of the integrated safety analysis (ISA), and the determination of whether an item relied on for safety (IROFS) has special or unique safety significance".

- NUREG-0700 (NRC 2002), covers many elements of HFE and their guidelines. The document groups these HFE elements into the following areas:

  1) *Information Displays* – general and specific guidelines, format elements, navigation, data quality, and update rates.
  2) *User-Interface Interaction and Management* – protection of data integrity (e.g., setpoints and process parameters) while still providing useful communication and control with the operator.
  3) *Analog Control and Display Devices* – hard wired display control devices, such as meters, pushbuttons, relays, potentiometers, and other non-digital controls.
  4) *Alarm Systems* - guidelines address the selection of alarm conditions, setpoints, processing, availability, display information, and controls.
  5) *Safety Parameter Display System* - displays of critical safety functions and parameters.
  6) *Large Panel HSI Systems* - presents information to multiple individuals while at their workstations or moving within a process area or control room.
  7) *Soft Control Systems* – operator aids mediated by software instead of direct physical input and output connections with a device.
  8) *Computer-Based Procedure Systems* – training tool to conveniently provide procedural-based information (versus multiple hard copies) that is up to date and controlled.
  9) *Automation Systems* - aids provided to personnel for situation analysis and decision making.

10) *Communication Systems* - preparing, addressing, transmitting, and receiving messages

11) *Workstation Design* - workstation features such as control-display integration and layout, labeling, and ergonomics

12) *Workplace Design* - overall layout of the workstations, other equipment, group-view displays, provisions for work equipment storage, and tracking environmental properties (e.g., noise, temp, ventilation flow, etc.)

13) *Maintainability of Digital Systems* - task support, equipment access, and support software and hardware for maintenance actions.

14) *Degraded HSI and Instrumentation and Control (I&C) Conditions* - HSI and I&C degradations and failures, such as alarms, displays, support systems, and controls.

15) *Integration of HSI Resources* – where the HSI design supports effective (1) monitoring, detection, and situation assessment; (2) response planning and implementation; and (3) team processes.

- NUREG-0711 (NRC 2012). Mentions the defense-in-depth contributions of the HFI/HFE to control hazards and provides several examples of contributing programs that support this control.

  "*Plant safety requires "defense in depth" that encompasses using multiple barriers to prevent the release of radioactive materials and employs a variety of programs to assure the integrity of barriers and related systems. These programs include conservative design, quality assurance, administrative controls, and human factors. Human factors engineering (HFE) plays a major role in supporting plant safety and providing defense in depth.*"

3.5.2 Concept of Operations

The design of the plant I&C systems utilized to perform an IROFS function and the HSI consider the concept of operations including (1) the physical characteristics and technical abilities of the operating staff, (2) shift staffing and organization, and (3) responsibilities of the operational staff.

A description of the operational concepts and assumptions relative to the staffing, personal characteristics, division of team responsibilities, and other related issues that form the basis for the HSI design.

The concept of operations is primarily concerned with the operating team. The secondary concern includes system users to participate in the design of user interfaces. Risk levels due to failures of administrative controls are identified within ISAS §2, Tables 2-5 and 2-6. ISAS Attachment 1 discusses the accident sequences and the assignment of failure indices for the administrative failures to ensure adequate levels of protection are provided to control each sequence.

3.5.3 Functional Requirements Specification

Functional requirements will be included in design documents for the HSI to address the concept of operation, personnel functions and tasks that support their role in the plant as derived from function, task, and staffing/qualifications analyses, and personnel requirements for a safe, comfortable working environment. Requirements will be established for various types of HSIs,

e.g., alarms, displays, and controls. Probability of Failure on Demand and Faults Trees are two potential methods to evaluate the failure probabilities of integrated systems that contain HSI.

3.5.4 HSI Concept Design

OE implements a modern I&C design utilizing experience gained at the Georges Besse II Plant. The HSI approach will utilize I&C concepts discussed in multiple standards and guidance documents. Examples of these are given by the following:

- ANSI/ISA-101.01-2015, *Human Machine Interfaces for Process Automation Systems*
- ANSI B11.26-2024, *Functional Safety for Equipment: General Principles for the Design of Safety Control Systems Using ISO 13849-1*
- ANSI/ISO 13849-1:2023, Safety *of machinery - Safety-related parts of control systems - Part 1: General principles for design*
- ANSI/ISO 13849-2:2012, *Safety of machinery - Safety-related parts of control systems - Part 2: Validation*
- ANSI/ISO 6385:2016, *Ergonomics principles in the design of work systems*

3.5.5 HSI Detailed Design and Integration

A style guide will be developed for use in the design of HSI features, layout, and environment. The content of the style guide will be derived from (1) the application of generic Human Factors Engineering (HFE) guidance and (2) guidance developed from design-related analyses and experience. The style guide supports the interpretation and comprehension of design guidance and helps to maintain consistency in the design across the HSIs. The primary topics addressed by the style guide include data presentation, screen-based data presentation, hierarchy, and navigation, presentation and operation of controls, and presentation and interpretation of alarms.

3.5.6 HSI Tests and Evaluations (Verification and Validation)

Verification and validation of the HSI design should be performed so that the as-built HSIs (1) are complete and operable, (2) conform to standard HFE principles and requirements, (3) are free of safety issues and human performance issues, and (4) implement the design accurately in the final design output documentation.

Testing and evaluation should be conducted throughout the HSI development process. Activities such as concept testing, mock-up activities, trade-off evaluations, and performance-based tests may be utilized at various stages of the design.

3.5.7 HSI Design Documentation

The HSI designs are documented using specific control process requirements. The various configuration management, design change controls, design verification, and design quality control tools are described in the OE Quality Assurance Program Description (QAPD). Configuration Management through the Engineering design change control process, and the Nuclear Safety review of design and process changes based on 10 CFR 70.72 requirements and Regulatory Guide 3.74 (NRC 2012).

3.6 REFERENCES

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NRC. 2015. *NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility.* Washington, D.C.: U.S. Nuclear Regulatory Commission (NRC).

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Chapter 4 – Radiation Protection

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4.0 RADIATION PROTECTION

This Chapter describes the facility Radiation Protection Program of the OE Facility. The Radiation Protection Program protects the radiological health and safety of workers and complies with the regulatory requirements in 10 CFR 19, 20, 30, 40 and 70.

4.1 COMMITMENT TO RADIATION-PROTECTION PROGRAM IMPLEMENTATION

The Radiation Protection Program meets the requirements of 10 CFR 20 Subpart B - Radiation Protection Programs (20.1101 (a)-(d)) and is consistent with the guidance provided in Regulatory Guide 8.2, Administrative Practices in Radiation Survey and Monitoring (NRC 2011a). OE management develops, documents and implements its Radiation Protection Program commensurate with the risks posed by a uranium enrichment operation. OE management uses, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA). The Radiation Protection Program content and implementation are reviewed at least annually as required by 10 CFR 20.1101(c). In addition, in accordance with 10 CFR 20.1101(d), constraints on atmospheric releases are established for the OE Facility such that no member of the public would be expected to receive a total effective dose equivalent in excess of 0.1 mSv/yr (10 mrem/yr) from these releases. Additional information regarding compliance with 10 CFR 20.1101(d) is provided in Chapter 9.

The Radiation Protection Manager 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 Radiation Protection Program, if warranted, are initiated and processed by the Radiation Protection Manager as part of the annual review process. Any resulting changes to the Radiation Worker Training module are also implemented.

The facility's administrative personnel exposure limits are set below the limits specified in 10 CFR Part 20. This provides assurance that legal radiation exposure limits are not exceeded and that the ALARA principle is emphasized. The facility administrative exposure limits are given in Table 4-1. Estimates of the facility area radiation dose rates and individual personnel exposures, during normal operations, are shown in Table 4-2 and Table 4-3. These estimates are based upon the operating experience of similar facilities in Europe.

Annual whole-body dose equivalents accrued by workers at an operating uranium enrichment plant are typically low. The maximum individual annual dose equivalents for the years 2021 through 2023 at the Orano Tricastin site, located in France, are summarized in Table 4-4. The worker maximum and average doses are steady over this time period. To put these doses in perspective, note that in the United States, individuals receive an annual effective dose equivalent of approximately 3.0 mSv (300 mrem) from background radiation (NCRP 1987).

4.1.1 Responsibilities of Key Program Personnel

In this section the organizational structure of the Radiation Protection Program is described, along with the responsibilities of key personnel. These personnel play an important role in the protection of workers, the environment and implementation of the ALARA program. Chapter 2 discusses the facility organization and administration in further detail, including a detailed discussion of the responsibilities of key management personnel.

4.1.2 Staffing of the Radiation Protection Program

Only suitably trained radiation protection personnel are employed at the facility. Members of the Radiation Protection Program staff are trained and qualified consistent with the guidance provided in American National Standards Institute (ANSI) standard 3.1 (ANSI 2020).

Sufficient resources in terms of staffing and adequate facilities, equipment and procedures are provided to implement an effective Radiation Protection Program.

Table 4-1 Occupational Administrative Radiation Exposure Limits

	Administrative Limit
Total Effective Dose Equivalent (TEDE)	10 mSv/yr (1,000 mrem/year)

Notes:

1. Excludes accident situations.
2. No routine extremity or skin monitoring is required.
3. TEDE is the sum of internal dose and external dose received during routine operations.
4. The Administrative Limit represents 20% of the NRC limit of 50 mSv/yr (5,000 mrem/yr) given in 10 CFR 20.1201.

Table 4-2 Estimated Dose Rates

Area or Component	Dose Rate, mSv/hr (mrem/hr)
Plant General Area (excluding Separation Building Modules)	<0.0001 (<0.01)
Separation Building Modules	0.001 (0.1)
Empty Used UF_6 Shipping Cylinder	0.1 on contact (10) 0.01 at 1 meter (1)
Full UF_6 Shipping Cylinder	0.05 on contact (5) 0.002 at 1 meter (0.2)

Table 4-3 Estimated Individual Exposures

Position	Annual Dose, mSv (mrem)
General Office Staff	< 0.05 (< 5)
Typical Operations & Maintenance Technician	1 (100)
Typical Cylinder Handler	3 (300)

Table 4-4 Annual Maximum and Average Worker Doses at Tricastin

Year	Maximum Annual Worker Dose Equivalent, mSv (mrem)	Average Annual Worker Dose Equivalent, mSv (mrem)
2021	0.926 (92.6)	0.162 (16.2)
2022	0.819 (81.9)	0.152 (15.2)
2023	0.846 (84.6)	0.169 (16.9)

Source: Orano internal document « Bilan Annuel de sureté de l'INB 168 année 2023

4.2 COMMITMENT TO AN ALARA PROGRAM

The objective of the ALARA program is to make every reasonable effort to maintain facility exposures to radiation as far below the dose limits of 10 CFR 20.1201 as is practical and to maintain radiation exposures to members of the public such that they are not expected to exceed the dose constraints of 10 CFR 20.1101(d). The design and implementation of the ALARA program is consistent with the guidance provided in Regulatory Guides 8.2 (NRC 2011a), 8.13 (NRC 1999a), 8.29 (NRC 1996), and 8.37 (NRC 1993g). The operation of the facility is consistent with the guidance provided in Regulatory Guide 8.10 (NRC 2016). The guidance of Regulatory Guide 4.21 will be followed to minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste (NRC 2008).

Annual doses to individual personnel are maintained ALARA. In addition, the annual collective dose to personnel (i.e., the sum of all annual individual doses, expressed in person-Sv or person-rem) is maintained ALARA. The dose equivalent to the embryo/fetus is maintained below the limits of 10 CFR 20.1208.

The Radiation Protection Program is written and implemented to ensure that it is comprehensive and effective. The written program documents policies that are implemented to ensure the ALARA goal is met. Facility procedures are written so that they incorporate the ALARA philosophy into the routine operations of the facility and ensure that exposures are consistent with 10 CFR 20.1101 limits. As discussed in Section 4.7, radiological zones will be established within the facility. The establishment of these zones supports the ALARA commitment in that the zones minimize the spread of contamination and reduce unnecessary exposure of personnel to radiation.

Specific goals of the ALARA program include maintaining occupational exposures as well as environmental releases as far below regulatory limits as is reasonably achievable. The ALARA concept is also incorporated into the design of the facility. The size and number of areas with higher doses rates are minimized consistent with accessibility for performing necessary services in the areas. Areas where facility personnel spend significant amounts of time are designed to maintain the lowest dose rates reasonably achievable.

4.3 ORGANIZATION AND PERSONNEL QUALIFICATIONS

The regulation 10 CFR 70.22 requires that the technical qualifications, including training and experience of facility staff, be provided in the license application. This information is provided in this section.

The Radiation Protection Program staff is assigned responsibility for implementation of the Radiation Protection Program functions. Only suitably trained radiation protection personnel are employed at the facility. Staffing is consistent with the guidance provided in Regulatory Guides 8.2 (NRC 2011a) and 8.10 (NRC 2016).

The Industrial Safety and Radiation Protection Manager's qualification requirements are described in Chapter 2.

As stated in Section 4.1.2 other members of the Radiation Protection Program staff are trained and qualified consistent with the guidance provided in ANSI standard 3.1 (ANSI 2020).

4.4 COMMITMENT TO WRITTEN PROCEDURES

OE will prepare written radiation protection procedures and Radiation Work Permit (RWP) to comply with requirements by 10 CFR 70.22(8)(a):

- Prepare written, approved procedures to carry out activities related to the radiation protection program. Procedures should address applicable radiation protection requirements found in 10 CFR 19, 20, 70, and 71 and any other applicable regulations.
- Establish a process for procedure generation or modification, authorization, distribution, and training, such that changes in technology or practices are communicated effectively and in a timely manner. Review and revise procedures, as necessary, to incorporate any facility or operational changes, including changes in the ISA. The radiation safety officer, or an individual who has the qualifications of the radiation safety officer, should approve all procedures related to radiation protection.
- Specify written, approved RWPs for activities involving licensed material that are not covered by written radiation protection procedures. RWPs should define the authorized activities, the level of approval required (a radiation specialist, as a minimum), information requirements, period of validity, expiration and termination times, and recordkeeping requirements.

Chapter 11 describes the program implemented for the control of procedures.

4.4.1 Radiation Work Permit Procedures

All work performed in Restricted Areas is performed in accordance with an RWP. The procedures controlling RWPs are consistent with the guidance provided in Regulatory Guide 8.10 (NRC 2016). An RWP may also be required whenever the Industrial Safety and Radiation Protection Manager deems that one is necessary. Activities involving licensed materials not covered by operating procedures and where radioactivity levels are likely to exceed airborne radioactivity limits require the issuance of an RWP. Both routine and non-routine activities are performed under an RWP. The RWP provides a description of the work to be performed. That is, the RWP defines the authorized activities. The RWP summarizes the results of recent dose rate surveys, contamination surveys, airborne radioactivity results, etc. The RWP specifies the precautions to be taken by those performing the task. The specified precautions may include personal protective equipment to be worn while working (e.g., gloves, respirators, personnel monitoring devices), stay-times or dose limits for work in the area, record keeping requirements (e.g., time or dose spent on job) and the attendance of a radiation protection technician during the work. The RWP requires approval by the Industrial Safety and Radiation Protection Manager or designee. The designee must meet the requirements of Section 4.1.2. RWPs have a predetermined period of validity with a specified expiration or termination time.

Standing RWPs are issued for routinely performed activities, such as tours of the plant by shift personnel or the changing of cylinders. A Standing RWP would, for example, be used for the job evolution of cylinder changing; a new RWP is not issued each time a new cylinder is changed.

Listed below are requirements of the Radiation Work Permit procedures:

- The Industrial Safety and Radiation Protection Manager or designee is responsible for determining the need for, issuing and closing out RWPs.
- Planned activities or changes to activities inside Restricted Areas or work with licensed materials are reviewed by the Industrial Safety and Radiation Protection Manager or designee for the potential to cause radiation exposures to exceed action levels or to produce radioactive contamination.
- RWP's include requirements for any necessary safety controls, personnel monitoring devices, protective clothing, respiratory protective equipment, and air sampling equipment and the attendance of radiation protection technicians at the work location.
- RWP's are posted at access points to Restricted Areas with copies of current RWP's posted at the work area location.
- RWP's clearly define and limit the work activities to which they apply. A RWP is closed out when the applicable work activity for which it was written is completed and terminated.
- RWP's are retained as a record at least for the life of the facility.

4.5 RADIATION SAFETY TRAINING

The design and implementation of the radiation protection training program comply with the requirements of 10 CFR 19.11 and 19.12. Records are maintained in accordance with 10 CFR 20.2110.

The radiation protection training program takes into consideration a worker normally assigned work activities. Abnormal situations involving exposure to radiation and radioactive material, which can reasonably be expected to occur during the life of the facility, are also evaluated and factored into the training. The extent of these instructions is commensurate with the potential radiological health protection problems present in the workplace.

Retraining of personnel previously trained is performed for radiological, chemical, industrial, and criticality safety at least annually. The retraining program also includes procedure changes and updating and changes in required skills. Changes to training are implemented, when required, due to incidents potentially compromising safety or if changes are made to the facility or processes. Records of training are maintained in accordance with the OE records management system. Training programs are established in accordance with Chapter 11. The radiation protection sections of the training program are evaluated at least every 3 years. The program content is reviewed to ensure it remains current and adequate to assure worker safety.

Radiation protection training is highlighted to emphasize the high level of importance placed on the radiological safety of plant personnel and the public. In-depth radiation protection training is provided for the various types of job functions (e.g., production operator, radiation protection technician, contractor personnel) commensurate with the radiation safety responsibilities associated with each such position. Visitors to a Restricted Area are trained in the formal training program or are escorted by trained personnel while in the Restricted Area.

Personnel access procedures ensure the completion of formal nuclear safety training prior to permitting unescorted access into the Restricted Areas. Training sessions covering criticality

safety, radiation protection and emergency procedures are conducted on a regular basis to accommodate new employees or those requiring retraining. Retraining is conducted when necessary to address changes in policies, procedures, requirements and the ISA.

Specific topics covered in the training program are listed in Chapter 11. The training provided includes the requirements of 10 CFR 19.

Individuals attending these sessions must pass an initial examination covering the training contents to assure the understanding and effectiveness of the training. The effectiveness and adequacy of the training program curriculum and instructors are also evaluated by audits performed by operational area personnel responsible for criticality safety and radiation protection.

Since contractor employees may perform diverse tasks in the Restricted Areas or Controlled Areas of the facility, formal training for these employees is designed to address the type of work they perform. In addition to applicable radiation safety topics, training contents may include RWPs, special bioassay sampling, and special precautions for welding, cutting, and grinding. Instructors certified by the Industrial Safety and Radiation Protection Manager conduct the radiation protection training programs.

The Industrial Safety and Radiation Protection Manager is responsible for establishing and maintaining the radiation protection training for all personnel, including contractor personnel who may be working at the facility. Records are maintained by the Training Manager for each employee documenting the training date, scope of the training, identity of the trainer(s), any test results and other associated information.

Individuals requiring unescorted access to a Restricted Area receive annual retraining. Contents of the formal radiation protection training program are reviewed and updated as required at least annually to ensure that the programs are current and adequate.

4.6 VENTILATION AND RESPIRATORY PROTECTION PROGRAMS

The regulations contained in 10 CFR 20, Subpart H, define the required elements of the facility respiratory protection and ventilation programs. This section describes the design and management measures taken to ensure that the installed ventilation and containment systems operate effectively. This section also describes the worker respiratory protection program.

4.6.1 Ventilation Program

The confinement of uranium and the attenuation of its associated radiation are a design requirement for the facility. The internal radiation exposure of workers is controlled primarily by the containment of Uranium Hexafluoride (UF_6) within process equipment. The entire UF_6 enrichment process, except for liquid sampling, is operated under a partial vacuum so that leaks are into the system and not into work areas.

Ventilation systems for the various buildings control the temperature and the humidity of the air inside the building. The ventilation systems serving normally non-contaminated areas exhaust approximately 10% of the air handled to the atmosphere. Ventilation systems serving potentially contaminated areas include design features that provide for confinement of radiological contamination. Ventilation systems for potentially contaminated areas (e.g., the Ventilated Room and Decontamination Workshop) exhaust 100% of the air handled to the environment through the exhaust vents. The systems contain filters to remove radioactive materials from the gas stream prior to release from the plant. Continuous Hydrogen Fluoride (HF) monitors are provided upstream of the filters with high level alarms to inform operators of UF_6 releases in the plant.

Normal operation of the facility will not result in a release of radioactive material that exceeds regulatory limits. Ventilation systems for areas that do not have the potential for contamination are not monitored for radioactivity because radioactive material is not handled or processed in these areas. No emergency ventilation systems are provided for operation when the normal ventilation systems are shut down.

Several measures are in place to ensure effective operation of the ventilation systems. Differential pressure across High Efficiency Particulate Air (HEPA) filters in potentially contaminated ventilation exhaust systems is monitored monthly or automatically monitored and alarmed. Operating procedures specify limits and setpoints on the differential pressure consistent with manufacturers' recommendations. Filters are changed if they fail to function properly or if the differential pressure exceeds the manufacturers' ratings.

Filter inspection, testing, maintenance and change out criteria are specified in written procedures. Change out frequency is based on considerations of filter loading, operating experience, differential pressure data and any UF_6 releases indicated by HF alarms.

Air flow rates at exhausted enclosures and close-capture points, when in use, are adequate to preclude escape of airborne uranium and minimize the potential for intake by workers. Air flow rates are checked monthly when in use and after modification of any hood, exhausted enclosure, close-capture point equipment or ventilation system serving these barriers. The various programs that pertain to preventive and corrective maintenance are described in Chapter 11, Sections 11.2.2 and 11.2.3 respectively.

Ventilation and containment systems (serving contaminated and potentially contaminated areas of the facility) will be designed and sized appropriately to reduce airborne concentrations below the occupational, derived air concentration (DAC) values specified in 10 CFR 20, Appendix B, during normal operations.

4.6.2 Respiratory Protection Program

The Facility uses process and engineering controls to control the concentration of radioactive material in air. However, there may be instances when it is not practical to apply process or other engineering controls. When it is not possible to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area, other means are implemented to maintain the total effective dose equivalent ALARA. In these cases, the ALARA goal is met by an increase in monitoring and the limitation of intakes by one or more of the following means:

- Control of access
- Limitation of exposure times
- Use of respiratory protection equipment
- Other controls, as available and appropriate
- If an ALARA analysis is performed to determine whether or not respirators should be used, safety factors other than radiological factors may be considered. The impact of respirator use on workers' industrial health and safety is factored into decisions to use respirators.

The respiratory protection program is developed to meet the requirements of 10 CFR 20, Subpart H.

Records of the respiratory protection program (including training for respirator use and maintenance) are maintained in accordance with the facility records management program as

described in Chapter 11. Respiratory protection procedures are revised as necessary whenever changes are made to the Facility, processing or equipment.

4.7 RADIATION SURVEYS AND MONITORING PROGRAMS

The Radiation Surveys and Monitoring Programs are based on the requirements of 10 CFR 20 Subpart F and ALARA principles. Written procedures are prepared for the elements of the Radiation Surveys and Monitoring Programs in compliance with the requirements of 10 CFR 20 Subpart F, Subpart C, Subpart L, and Subpart M. The Reporting Program is consistent with the requirements of 10 CFR Part 19 and 10 CFR Part 20.

All personnel who enter Restricted Areas (as defined below) are required to wear personnel monitoring devices that are supplied by a vendor that holds dosimetry accreditation from the National Voluntary Laboratory Accreditation Program. In addition, personnel are required to monitor themselves prior to exiting Restricted Areas which may have the potential for contamination.

Continuous airborne radioactivity monitors provide indication of the airborne activity levels in the Restricted Areas of the Facility. Monitoring instruments for airborne alpha emitters are provided at different locations throughout the Facility. These monitors are designed to detect alpha emitters in the air, which would indicate the potential for uranium contamination. When deemed necessary, portable air samplers may be used to collect a sample on filter paper for subsequent analysis in the laboratory.

Monitor data is collected for regular analysis and documentation. Monitors in locations classified as Airborne Radioactivity Areas are equipped with alarms. The alarm is activated when airborne radioactivity levels exceed predetermined limits. The limits are set with consideration being given to both toxicity and radioactivity. The volume of air sampled may have to be adjusted to ensure adequate sensitivity with minimum sampling time. The operating history of the facility, changes in technology, changes in room functions and design, and changes in regulations may necessitate adjustment of the monitors.

Continuous monitoring of direct radiation exposure rates is not performed because the uranium processed in the facility is handled in closed containers. The radionuclides of interest are primarily alpha and beta emitters. The decay data for these radionuclides are shown in Table 4-5, Radiation Emitted from Natural UF_6 Feed.

Alpha and beta radiation cannot penetrate the container walls. Typical area radiation monitors measure gamma radiation. At the Facility, the gamma radiation is not present at sufficient levels to provide representative indications. Instead, periodic radiation monitoring is performed with portable survey meters and "wipe tests" for contaminations are taken to evaluate radiological conditions in the facility.

A calibration is performed in accordance with written established procedures and documented prior to the initial use of each airflow measurement instrument (used to measure flow rates for air or effluent sampling) and each radioactivity measurement instrument. Periodic operability checks are performed in accordance with written established procedures. Calibrations are performed and documented on each airflow measurement and radioactivity measurement instrument at least annually (or according to manufacturers' recommendations, whichever is more frequent) or after failing an operability check, or after modifications or repairs to the instrument that could affect its proper response, or when it is believed that the instrument has been damaged.

Unreliable instruments are removed from service until repairs are completed. Portal monitors, hand and foot monitors and friskers have the required sensitivity to detect alpha contamination

on personnel to ensure that radioactive materials do not spread to the areas outside the Restricted Areas.

The background and efficiency of laboratory counting instruments, when used for radiation protection purposes, is determined daily. This determination may be less frequent only, if necessary, due to long counting intervals.

4.7.1 Radiological Zones

Radiological zones within the facility have been established to (1) control the spread of contamination, (2) control personnel access to avoid unnecessary exposure of personnel to radiation, and (3) to control access to radioactive sources present in the facility.

Table 4-2 lists general dose rate estimates for the facility. These dose estimates were prepared based upon historical data from similar operating centrifuge enrichment facilities. Areas associated with higher dose rates may be restricted from public access, as determined by facility management. Areas where facility personnel spend substantial amounts of time are designed to minimize the exposure received when routine tasks are performed, in accordance with the ALARA principle.

The following definitions of areas are provided to describe how the Facility Radiation Protection Program is implemented to protect workers and the general public on the site.

4.7.1.1 Unrestricted Area

NRC regulation 10 CFR 20.1003 defines an unrestricted area as an area, access to which is neither limited nor controlled by the licensee. The area adjacent to the Facility site where the Facility does not normally exercise access control is an Unrestricted Area. This area can be accessed by members of the public, indigenous wildlife, or by facility personnel. The Unrestricted Area is governed by the limits in 10 CFR 20.1301. The total effective dose equivalent to individual members of the public from the licensed operation may not exceed 1 mSv (100 mrem) in a year (exclusive of background radiation). The dose in any Unrestricted Area from external sources may not exceed 0.02 mSv (2 mrem) in any one hour. In addition to the NRC limit, the Environmental Protection Agency, in 40 CFR 190, imposes annual dose equivalent limits of 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid, and 0.25 mSv (25 mrem) to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials to the general environment from uranium fuel cycle operations and to radiation from these operations.

4.7.1.2 Restricted Area

The NRC defines a restricted area as an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Access to and egress from a Restricted Area at the plant site is through a radiation protection control point known as a Monitor Station. Monitoring equipment is located at these egress points to ensure that equipment and materials removed from restricted areas to unrestricted areas are not contaminated above the release levels presented in Appendix A, “Acceptable Surface Contamination Levels,” to Regulatory Guide 8.24. All personnel are required to monitor themselves prior to exiting Restricted Areas that have the potential for contamination, using monitoring instruments that detect gross alpha contamination.

Examples of Restricted Areas include storage areas for UF_6 in the Cylinder Receipt and Shipping Building and the potentially contaminated areas in the Technical Support Building. Personnel who

have not been trained in radiation protection procedures are not allowed to access a Restricted Area without escort by trained personnel.

The areas defined below may exist within a Restricted Area. These areas may be temporary or permanent. The areas are posted to inform workers of the potential hazard in the area and to help prevent the spread of contamination. These areas are conspicuously posted in accordance with the requirements of 10 CFR 20.1902.

- An area in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.05 mSv (0.005 rem) "in 1 hour at 30 centimeters" from the radiation source or from any surface that the radiation penetrates is designated a "Radiation Area" as defined in 10 CFR 20.1003.
- As defined in 10 CFR 20.1003, "Airborne Radioactivity Area" means a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations: (1) In excess of the DACs specified in appendix B to 10 CFR 20.1001 - 20.2401, or (2) To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the annual limit on intake (ALI) or 12 DAC-hours. Note that entry into this area does not automatically require the wearing of a respirator.
- A "High Radiation Area" as defined in 10 CFR 20.1003, is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1 mSv (0.1 rem) "in 1 hour at 30 centimeters" from the radiation source or from any surface that the radiation penetrates. No examples of this type of area are expected during routine operation of the facility. This designation is provided here only for the purposes of emergency situations (drills and actual).
- The OE Facility defines a "Contaminated Area" as an area where removable contamination levels are greater than 20 dpm/100 cm^2 of alpha activity or 1,000 dpm/100 cm^2 beta/gamma activity.

The NRC limits the soluble uranium intake of an individual to 10 milligrams in a week in consideration of chemical toxicity. The Facility posts areas where the intake of soluble uranium in one week is likely to exceed 1 milligram if respiratory protection is not utilized.

4.7.1.3 Controlled Area

In 10 CFR 20.1003, the NRC defines a "Controlled Area" as an area, outside of a Restricted Area but inside the site boundary, access to which can be limited by the licensee for any reason. The area of the plant within the perimeter fence but outside any Restricted Area is part of the Controlled Area. Due to the presence of the owner-controlled area fence, members of the public do not have direct access to this Controlled Area of the site and must be processed by security and authorized to enter the site. Training for access to a Controlled Area is provided commensurate with the radiological hazard.

Site visitors include delivery people, tour guests and service personnel who are temporary, transient occupants of the Controlled Area. Area monitoring demonstrates compliance with public exposure limits for such visitors. All individuals who are contractor or OE employees and who work only in the Controlled Area are subject to the exposure limits for members of the public as stated in 10 CFR 20.1301.

4.7.2 Access and Egress Control

The facility establishes and implements an access control program that ensures that (a) signs, labels, and other access controls are properly posted and operative, (b) restricted areas are established to prevent the spread of contamination and are identified with appropriate signs, and (c) step-off pads, change facilities, protective clothing facilities, and personnel monitoring instruments are provided in sufficient quantities and locations.

Because there are no High Radiation Areas in the facility, there are no areas where access is physically prevented due to radiation level. Access control is by administrative methods. Access to certain areas may be physically prevented for security reasons. Personnel who have not been trained in radiation protection procedures are not allowed access to a Restricted Area without escort by other trained personnel.

Access to and egress from a Restricted Area is through one of the monitor stations at the particular Restricted Area boundary. Access to and egress from each Radiation Area, Contaminated Area or Airborne Radioactivity Area within a Restricted Area may also be individually controlled. A contamination monitor (e.g., frisker, hand and foot monitor or portal monitor, step-off pad and container for any discarded protective clothing may be provided at the egress point from certain of these areas to prevent the spread of contamination.

4.7.3 Posting for Radiation Protection Awareness

Restricted Areas and other areas within the Restricted Areas (e.g., Airborne Radioactivity Area) are clearly identified by physical means such as placarding or boundary marking, so that facility personnel can identify these areas and use their training to minimize their exposure. This identification is done in accordance with 10 CFR 20.1902. The radiation and contamination levels from the most recent survey are clearly noted on each posting.

4.7.4 Protective Clothing and Equipment

The proper use of protective clothing and equipment can minimize internal and external exposures to radioactivity. Personnel working in areas that are classified as Airborne Radioactivity Areas or in Contaminated Areas must wear appropriate protective clothing. Areas requiring protective clothing are posted at each of their entry points.

Radiation Protection Program staff are responsible for determining the need for protective clothing in each work area.

4.7.5 Personnel Monitoring for External Exposures

External exposures are received primarily from the radioactive decay products of ^{235}U and ^{238}U contained in majority in the UF_6 cylinders. Over the life of the Facility, the number of full depleted uranium (DUF_6) cylinders placed on the storage pads may increase to the pads' design capacity. As a result, it is possible that the neutron contribution to the total worker dose may require monitoring. The neutrons are due to spontaneous fission in uranium as well as the alpha, neutron reaction on fluorine. Workers receive training regarding ALARA concepts such as time-distance-shielding to minimize their exposures.

All personnel whose duties require them to enter Restricted Areas wear individual external dosimetry devices, e.g., passive dosimeters such as thermoluminescent dosimeters (TLDs) that are sensitive to beta, gamma and neutron radiation. External dosimetry devices are evaluated at

least quarterly to ascertain external exposures. Administrative limits on radiation exposure are provided in Table 4-1 .

If the pro-rated radiation exposure administrative limit set in the Radiation Protection Program is exceeded in any quarter, then an investigation is performed and documented to determine what types of activities may have contributed to the worker's external exposure.

Anytime an administrative limit is exceeded, the Industrial Safety and Radiation Protection Manager is informed. The Industrial Safety and Radiation Protection Manager is responsible for determining the need for and recommending investigations or corrective actions to the responsible Manager(s). Copies of the Industrial Safety and Radiation Protection Manager's recommendations are provided to the Safety Review Committee.

4.7.6 Personnel Monitoring for Internal Exposures

Internal exposures for all personnel wearing external dosimetry devices are evaluated via direct bioassay (e.g. in vivo body counting), indirect bioassay (e.g., urinalysis), or an equivalent technique, at least annually.

- For soluble (Class D) uranium, 10 CFR 20.1201(e) limits worker intake to no more than 10 milligrams (mg) of soluble uranium in a week.
- The Facility's annual administrative limit for the TEDE is 10 mSv (1.0 rem).

Continuous air monitoring in Airborne Radioactivity Areas may be performed to complement the bioassay program. Alarm setpoints on the continuous air monitors in the Airborne Radioactivity Areas may be used to provide an indication that internal exposures may be approaching the action limit.

If the Facility annual administrative limit is exceeded as determined from bioassay results, then an investigation is performed and documented to determine what types of activities may have contributed to the worker's internal exposure.

4.7.7 Evaluation of Doses

Dose evaluations may be performed at more frequent intervals and should be performed when reasonable suspicion exists regarding an abnormal exposure. The internal and external exposure values are summed in accordance with 10 CFR 20.1202. Procedures for the evaluation and summation of doses are based on the guidance contained in Regulatory Guides 8.7 (NRC 2018) and 8.34 (NRC 2022).

Table 4-5 Radiation Emitted from Natural UF_6 Feed

Element	Nuclide Symbol	Half-Life	Maximum Radiation Energies (Mev) and Intensities (%)		
			Alpha (α)	Beta (β)	Gamma (γ)
92 uranium	^{238}U	4.5E+9 years	4.15 25% 4.20 75%	None	0.013 8.8%
90 thorium	^{231}Th	26 hours	None	0.39 ~ 100%	0.025 14.7%
90 thorium	^{234}Th	24 days	None	0.19 73% 0 10 27%	0.06 3.8% 0.09 5.4%
91 protactinium	^{234m}Pa	1.2 minutes	None	2.28 99%	0.766 0.21% 1.001 0.60%
92 uranium	^{234}U	2.5E+5 years	4.72 28% 4.78 72%	None	0.053 0.12%
92 uranium	^{235}U	7.0E+8 years	4.37 17% 4.40 55% 4.60 14%	None	0.143 12% 0.185 54% 0.205 6%

4.8 ADDITIONAL PROGRAM COMMITMENTS

The following section describes additional program commitments related to the Radiation Protection Program.

4.8.1 Leak Testing Byproduct Material Sources

In addition to the uranium processed at the Facility, other sources of radioactivity are used. These sources are small calibration sources used for instrument calibration and response checking. These byproduct material sources may be in solid, liquid, or gaseous form; the sources may be sealed or unsealed. Both types of sources present a small radiation exposure risk to Facility workers. Typical byproduct material quantities and uses for a uranium enrichment centrifuge plant are summarized in Table 4-6. The byproduct materials for the Facility will be identified during the design phase, and the License Application will be revised accordingly. Leak-testing of sources is performed in accordance with the following NRC Branch Technical Positions:

- License Condition for Leak-Testing Sealed Byproduct Material Sources (NRC 1993b)
- License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters (NRC 1993c)
- License Condition for Leak-Testing Sealed Uranium Sources (NRC 1993d).

4.8.2 Records and Reports

The Facility Records Management program is described in Chapter 11. The Facility maintains complete records of the Radiation Protection Program for at least the life of the facility.

The Facility maintains records of the Radiation Protection Program (including program provisions, audits, and reviews of the program content and implementation), radiation survey results (air sampling, bioassays, external-exposure data from monitoring of individuals, internal intakes of radioactive material), and results of corrective action program referrals, RWPs and planned special exposures.

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

Table 4-6 Table Typical Quantities of Byproduct Material for a Uranium Enrichment Centrifuge Plant

Radionuclide	Quantity	Use
^{3}H	5.14E-01 Ci	Instrument calibration or response checking
^{36}Cl	2.26E-07 Ci	Instrument calibration or response checking
^{57}Co	2.51 E-02 Ci	Instrument calibration or response checking
^{90}Sr	2.81 E-08 Ci	Instrument calibration or response checking
^{99}Tc	8.35E-08 Ci	Instrument calibration or response checking
^{109}Cd	1.00E-03 Ci	Instrument calibration or response checking
^{131}Cs	1.05E-08 Ci	Instrument calibration or response checking
^{133}Ba	1.89E-05 Ci	Instrument calibration or response checking
^{137}Cs	5.53E-02 Ci	Instrument calibration or response checking
^{210}Po	1.70E-03 Ci	Instrument calibration or response checking
^{226}Ra	1.03E-03 Ci	Instrument calibration or response checking
^{233}U	1.00E-01 Ci	Instrument calibration or response checking
^{234}U	1.19E-10 Ci	Instrument calibration or response checking
^{235}U	1.00E-01 Ci	Instrument calibration or response checking
^{236}U	1.00E-01 Ci	Instrument calibration or response checking
^{237}Np	5.41 E-08 Ci	Instrument calibration or response checking
^{238}U	4.45E-09 Ci	Instrument calibration or response checking
^{241}Am	2.97E-02 Ci	Instrument calibration or response checking

Notes:

1. Byproduct material may be in solid, liquid, or gaseous form. Per Document Test (ANSI 2013)
2. Byproduct material is not necessarily restricted to sealed sources.

4.9 REFERENCES

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NRC. 1993c. "License Condition for Leak-Testing Sealed Sources Which Contain Alpha and/or Beta-Gamma Emitters." *U.S. Nuclear Regulatory Commission, April 1993* ADAMS ML17208A030, Branch Technical Position.

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NRC. 2022. "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses." *U.S. Nuclear Regulatory Commission, 2022* Regulatory Guide 8.34 (Revision 1).

NRC. 2016. "Operating Philosophy for Maintaining Occupational and Public Radiation Exposures As Low As Is Reasonably Achievable." *U.S. Nuclear Regulatory Commission, 2016* Regulatory Guide 8.10 (Revision 2).

Chapter 5 – Nuclear Criticality Safety

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5.0 NUCLEAR CRITICALITY SAFETY

5.1 USE OF INDUSTRY STANDARDS

The Nuclear Criticality Safety (NCS) Program for the Facility is in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.71 (NRC 2018), which provides guidance on complying with the applicable portions of NRC regulations This includes 10 CFR Part 70, by describing procedures for preventing nuclear criticality accidents in operations involving handling, processing, storing, and transporting special nuclear material (SNM) at fuel and material facilities.

OE is committed to following the guidelines in this regulatory guide for specific ANSI/ANS criticality safety standards:

- ANSI/ANS-8.1-2014 (ANSI 2023a)
- ANSI/ANS-8.3-2022 (ANSI 2022a)
- ANSI/ANS-8.24-2017 (ANSI 2023b)

5.2 CRITICALITY ACCIDENT ALARM SYSTEM

The Facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24. Each area where Special Nuclear Material (SNM) is handled, used, or stored are provided with CAAS coverage. The CAAS will be uniform throughout the Facility for the type of radiation detected and alarm signals. Documentation is maintained which demonstrates the CAAS meets the requirements of 10 CFR 70.24. Emergency management measures are covered in the Facility Emergency Plan.

The CAAS is designed, installed, and maintained in accordance with ANSI/ANS-8.3-2022 (ANSI 2022b) as modified by Regulatory Guide 3.71 (NRC 2018).

The CAAS is provided with emergency power and is designed to remain operational during credible events or conditions, including fire, explosion, corrosive atmosphere, or seismic shock.

Whenever the CAAS is not functional, compensatory measures, such as limiting access and restricting SNM movement, will be implemented. On-site guidance is provided and is based on process-specific considerations that consider applicable risk trade-off of the duration of reliance on compensatory measures versus the risk associated with process upset in shutdown.

5.3 EMERGENCY PLANNING AND RESPONSE

OE commits to ANSI/ANS-8.23-2019 (ANSI 2024b) for development of the Facility Emergency Plan (ANSI 2024b).

Pursuant to 10 CFR 70.24(b)(1), areas that require a CAAS are equipped with personnel accident dosimeters readily available with prompt onsite dosimeter readout.

Pursuant to 10 CFR 70.24(b)(2), the Facility includes arrangements for the on-site decontamination of personnel and the transport and medical treatment of exposed individuals outside the site boundary.

Additional information regarding nuclear accident planning and response is detailed in the Facility Emergency Plan.

5.4 SUBCRITICALITY AND DOUBLE-CONTINGENCY PRINCIPLE

The NCS criteria in Section 5.7 are used for managing criticality safety and include adherence to the double contingency principle as stated in the ANSI/ANS-8.1-2014 (ANSI 2023a). The adopted double contingency principle states:

> "*process design 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*."

Each process that has accident sequences that could result in an inadvertent nuclear criticality at the Facility meets the double contingency principle. To meet the double contingency principle, the Facility will incorporate into process designs sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Using these NCS criteria, including the double contingency principle, low enriched uranium enrichment facilities have never had an accidental criticality. The Facility will produce no greater than 10 wt.% enrichment. However, as additional conservatism, the nuclear criticality safety analyses are performed assuming a ^{235}U enrichment of 11 wt.%, except for Contingency Dump System upset events (both to the dump traps and to tails cylinders) which are analyzed assuming that the dumped material has a ^{235}U enrichment of 1.5 wt.%. Both the 10 wt.% value and the 1.5 wt.% value include appropriate margins for safety.

In accordance with 10 CFR 70.61(d), the general criticality safety philosophy is to prevent accidental uranium enrichment excesses, provide geometrical safety when practical, provide for moderation controls within the Uranium Hexafluoride (UF_6) processes and impose strict mass limits on containers of aqueous, solvent based, or acid solutions containing uranium.

All NCS controls are preventative in nature; there are no mitigative NCS controls. Interaction controls provide safe movement and storage of components. Facility and equipment features ensure the prevention of excessive enrichment.

As required by 10 CFR 70.64(a), by observing the double contingency principle throughout the Facility, a criticality accident is reduced.

5.5 ORGANIZATION AND ADMINISTRATION OF THE NCS PROGRAM

The NCS organization is responsible for implementing the Nuclear Criticality Safety Program.

The Nuclear Safety & Criticality Manager reports to the Environmental, Health, Safety, and Licensing (EHS&L) Manager and the organization is described in Chapter 2.

The designated responsibilities and minimum qualifications of the Nuclear Safety & Criticality Manager are described in Chapter 2.

The Nuclear Safety & Criticality Manager implements the NCS Program to meet the regulatory requirements of 10 CFR 70.

The NCS Program commits to the following objectives:

- Performing and documenting Nuclear Criticality Safety Evaluations (NCSE) for new or changed processes and establishing safety limits and controls as necessary to ensure that processes will remain subcritical under normal and credible abnormal conditions
- Establishing, as practicable, double-contingency protection and defense-in-depth measures; ensuring sufficient margins of safety and subcriticality to provide additional assurance that the likelihood of criticality will be acceptably low
- Establishing and maintaining a CAAS system and emergency-response procedures to protect health and safety in the event criticality occurs
- Providing technical support to emergency response personnel in responding to and recovering from abnormal conditions and emergencies up to and including a criticality accident
- Verifying the adequacy of criticality controls through audits and assessments, including observation of operations and verification of equipment configuration
- Ensuring the adequacy of NCSEs through peer reviews, self-assessments, and validation and verification of calculational methods
- Training and otherwise supporting operations in procedures to ensure the safe handling of special nuclear material
- Supporting regulatory compliance with regard to event reporting (10 CFR 70.50 and Appendix A to 10 CFR 70), complying with the facility change process (10 CFR 70.72), and participating in the performance and documentation of the facility's ISA (10 CFR 70.61 through 70.66) insofar as they pertain to criticality safety.

The NCS Program structure follows guidance in ANSI/ANS-8.1-2014 (ANSI 2023a) and ANSI/ANS-8.19-2014 (ANSI 2024a) including definition of the responsibilities and authorities of key program personnel will be provided.

The NCS emergency procedure training will be provided. The NCS postings will be provided and maintained current.

The NCS program will be used to establish and maintain NCS safety limits and NCS operating limits for IROFS in nuclear processes and a commitment to maintain adequate management measures to ensure the availability and reliability of the IROFS.

The NCS methodologies and technical practices will be kept applicable to current configuration by means of the configuration management function. The NCS program will be upgraded, as necessary, to reflect changes in the ISA or NCS methodologies and to modify operating and maintenance procedures in ways that could reduce the likelihood of the occurrence of an inadvertent nuclear criticality.

The NCS program will be used to evaluate modifications to operations, to recommend process parameter changes necessary to maintain the safe operation of the Facility, and to select appropriate IROFS and management measures.

The NCS program will be used to promptly detect NCS deficiencies by means of operational inspections, audits, and investigations. Deficiencies will be entered into the corrective action program to prevent the recurrence of unacceptable performance deficiencies in IROFS, NCS function, or management measures.

NCS program records will be retained as described in Chapter 11.

5.6 MANAGEMENT MEASURES APPLIED TO NCS PROGRAM

OE commits to following industry practices described in ANSI/ANS-8.19-2014 (ANSI 2024a) and ANSI/ANS-8.20-2025 (ANSI 2025) regarding training, procedures, and audits and assessments. In addition to the double contingency principle, effective management of the NCS Program including establishment of safety parameters and procedures. The NCS baseline design criteria requirements in 10 CFR 70.64(a) will be adhered to.

NCS Training will be provided to individuals who handle nuclear material at the facility. The training is based upon the training program described in ANSI/ANS-8.20-2025 (ANSI 2025). The training program is developed and implemented with input from the criticality safety staff, training staff, and management. The training focuses on the following:

- Appreciation of the physics of nuclear criticality safety.
- Analysis of jobs and tasks to determine what a worker must know to perform tasks efficiently.
- Design and development of learning objectives based upon the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker.
- Implementation of revised or temporary operating procedures.
- Required response to the activation of the CAAS.
- Required response to NCS nonconformance.

5.7 TECHNICAL PRACTICES FOR NCS

This section describes the methodologies and technical practices used to perform the NCSEs. The determination of the NCS controlled parameters and their application and the determination of the NCS limits on IROFS are also presented.

5.7.1 Calculational Method Validation

SCALE 6.2.4 (ORNL 2020) is a Monte Carlo tool for NCS analysis. The advanced geometry modeling capability and detailed continuous-energy and multigroup collision modeling treatments provide realistic 3- dimensional models for an accurate simulation of neutronic behavior to provide the best estimate neutron multiplication factor, k_{eff}. Complex models can be simply set up and verified. Additionally, SCALE 6.2.4 (ORNL 2020) has demonstrable accuracy over a wide range of applications and is distributed with a validation database comprising critical experiments covering uranium, plutonium and mixed systems over a wide range of moderation and reflection. The selected experiments are regarded as being representative of systems that are widely encountered in the nuclear industry, particularly with respect to chemical plant operations, transportation, and storage. The validation database is subject to on-going review and enhancement. A categorization option is available in SCALE 6.2.4 (ORNL 2020) to assist the criticality analyst in determining the type of system being assessed and provides a quick check that a calculation is adequately covered by validation cases.

The validation process establishes method bias by comparing measured results from laboratory critical experiments to method-calculated results for the same systems. The verification and validation processes are controlled and documented. The validation establishes a method bias

by correlating the results of critical experiments with results calculated for the same systems by the method being validated. Critical experiments are selected to be representative of the systems to be evaluated in specific design applications. The range of experimental conditions encompassed by a selected set of benchmark experiments establishes the area of applicability over which the calculated method bias is applicable. Benchmark experiments are selected that resemble as closely as practical the systems being evaluated in the design application.

The extensive validation database contains a number of solution experiments applicable to this application involving both low and high-enriched uranium. The SCALE 6.2.4 (ORNL 2020) code with the V7-252 group ENDF/B-VII.1 cross section library is validated against the experiments (OE 2025) provided in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (NEA 2023), NUREG/CR-1071 (NRC 1980), and benchmark catalogued in ORNL/CSD/TM-238 (ORNL 1986).

The validation process complies with the applicable requirements from ANSI/ANS-8.24-2017 (ANSI 2023b) as well as guidance from NUREG/CR-6698 (NUREG 2001), NUREG/CR-7311 (NUREG 2025) and NUREG-1520 (NRC 2015).

The criticality code validation methodology can be divided into four steps:

- Identify general Facility design applications (Area of Interest, AoI).
- Select applicable benchmark experiments for the Area of Applicability (AoA) that bounds the AoI.
- Model and calculate k_{eff} values of selected critical benchmark experiments.
- Perform statistical analysis of calculated results to determine computational bias and bias uncertainty associated with the computer code and cross section library to calculate an Upper Subcritical/Safety Limit, USL.

The AoA is established for the following four (4) parameters which are important for reactivity calculations: enrichment, moderation, epithermal fission fraction, calculation type and materials.

The area of applicability for enrichment includes benchmarks ranging from low-enriched material to mixtures of LEU and fissile material, up to systems containing 36.6 wt.% ^{235}U, which bound the Facility's maximum enrichment of 10 wt.% ^{235}U.

The area of applicability for moderation is supported by benchmarks spanning from low to high H/U-235 or C/U-235 atom-density ratio.

The area of applicability limits the contribution of epithermal-range fissions (0.8–10 eV, groups 145–209) to no more than 5% of total fissions for systems with moderation between $3 < H/X < 60$.

The area of applicability calculation type includes SCALE 6.2.4 (CSAS5 module) with the 252-group cross section library (v7-252). The following CSAS5 problem types are allowed: Infinite homogeneous, lattice cell, multi-region, and any geometry available in KENO V.a.

The area of applicability for material includes basic fissile material forms such as uranium oxide, uranium tetrafluoride, uranium metal, uranyl nitrate and uranyl fluoride.

A total of 601 experiments is selected for the validation effort; the description of all relevant experiments is provided in Appendix C of (OE 2025).

CSAS5 module of SCALE 6.2.4 is employed to model and calculate the k_{eff} ($k_{eff} = k_{KENO} + 2\,\sigma_{KENO}$) of the selected critical experiments. Appendix B of (OE 2025) presents the calculated results.

The statistical analysis of the selected critical experiments k_{eff} is then performed to determine the bias and USL.

The formulation for the USL, (NUREG 2025), is provided as:

$$USL = 1 + \beta - \sigma_{\beta} - M_A - M_D \text{ if } \beta \,\delta\, 0 \text{ or}$$

$$USL = 1 - \sigma_{\beta} - M_A - M_D \text{ if } \beta > 0$$

Where:

M_A is the administrative margin

M_D is the margin added to account for any deficiencies associated with the validation suite. M_D is referred to as Δ_{AoA} in (OE 2025). When no extrapolations are made to the AoA, then Δ_{AoA} is 0. This validation effort rigorously evaluates applicable benchmarks to ensure that the computational method is not extrapolated beyond its established AoA.

β is the bias and σ_{β} is the bias uncertainty.

When $\beta > 0$ then

$$USL = 1 - \sigma_{\beta} - M_A$$

NUREG-1520 Appendix B (NRC 2015) provides guidance in the development of adequate justification for the M_A (or Minimum Margin of Subcriticality, MMS, as referred to in NUREG-1520) based on the following approaches:

- Justification based on conservatism in calculational models
- Justification based on validation methodology and results
- Justification based on additional risk Informed considerations
- Justification based on statistical evaluation to establish MMS

NUREG-1520 (NRC 2015) Appendix B establishes that the first three approaches can provide an adequate technical basis for MMS selection, while the 4th method (based on statistical evaluation) is not considered by the NRC to be an appropriate basis for selection of a specific MMS.

Sections 2.3.1 through 2.3.3 of (OE 2025) address the 3 first justifications and provide the basis for using an MMS, or M_A, of 0.02 given the rigorous experiment selection and analysis and a conservative application of statistical methods to calculate bias, bias uncertainty, and the associated USL.

Table 5-1 summarizes the results of each USL calculation, for all cases and by subsets. The most limiting value is that of H/X: Epithermal ($3 < H/X < 60$) set which is 0.9525 (determined with 0.02 administrative margin).

SCALE 6.2.4, using the 252-group ENDF/B-VII cross-section library, has been validated for performing the criticality analyses at the Facility. Based on this validation, the calculated upper subcritical limit (USL) supports using a safety margin of 0.05 for nuclear criticality safety. This margin, expressed as $k_{eff} = k_{KENO} + 3\sigma_{KENO} < 0.95$, is sufficient to ensure that a system remains subcritical under normal and abnormal credible conditions. Therefore, the Facility will be designed based on:

$$k_{eff} = k_{KENO} + 3\sigma_{KENO} < 0.95$$

Table 5-1 SCALE-6.2.4 V7-252 Validation Summarv

5.7.2 Nuclear Criticality Safety Evaluations

OE is committed to performing NCSE in accordance with documented and approved administrative procedures adhering to these following principles:

a. NCS safety limits will be established based on analyses assuming optimum or the most reactive credible values of parameters unless specified controls are implemented to limit parameters to a particular range of values. When less than the optimum values are used, and corresponding controls are not identified, adequate justifications will be provided in the NCSE.
b. NCS operating limits maybe established as appropriate to ensure that safety limits are unlikely to be exceeded. If separate operating limits are specified, process variability and uncertainty should be considered.
c. The specific controls and management measures necessary to enforce NCSE safety limits and/or operating limits will be specified.

OE is committed to providing the technical basis that demonstrates subcriticality under normal and credible abnormal conditions and compliance with the double-contingency principle in the NCSE.

OE is committed to incorporating each CSE into its configuration-management program and its system of ISA documentation.

5.7.3 Evaluation and Implementation of Controlled Parameters

The use of a single criticality-safety control to maintain two or more parameters is credited as only one contingency for purposes of meeting the double-contingency principle.

OE commits to the preferred use of passive engineered controls; in particular, passive engineered geometry control.

OE commits to the following order of preference for NCS control: (1) passive engineered, (2) active engineered, (3) enhanced administrative, and (4) simple administrative controls.

OE commits to preference for designating explicit NCS controls over reliance on the natural and credible course of events.

OE commits to preference for control of two or more parameters over multiple controls on a single parameter. If relying on two or more controls on a single parameter. If relying on two or more controls on a single parameter, OE commits to preference for diverse over redundant means of control.

For the purposes of the NCSE, it is assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2. Most components that form part of the centrifuge plant or are connected to it reflect the assumption that any accumulation of uranium is taken to be in the form of a uranyl fluoride/water mixture at a maximum H/U atomic ratio of 7 (exceptions are discussed in the associated NCSE documentation). The ratio is based on the assumption that significant quantities of moderated uranium could only accumulate by reaction between UF_6 and moisture in air leaking into the plant process equipment. Due to the high vacuum requirements of a centrifuge plant, in-leakage is controlled at very low levels and thus the H/U ratio of 7 represents an abnormal condition. Gross water ingress into plant is not considered credible for the same reasons. In the

case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated PFPE (perfluorinated polyether) type lubricant. Mixtures of UF_6 and PFPE oil would be a less pessimistic case than the uranyl fluoride / water mixture considered since maximum HF solubility in PFPE is only approximately 0.1% by weight.

A maximum H/U ratio of 7 for the uranyl fluoride-water mixture is derived as follows. The stoichiometric reaction between UF_6 and water vapor in the presence of excess UF_6 can be represented by the equation:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound solid hydrates of compositions $UO_2F_2.1.5H_2O$ and $UO_2F_2.2H_2O$ can form in the presence of water vapor, the former composition being the stable form on exposure to atmosphere. Additionally, the HF produced by the UF_6 / water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:

$$UF_6 + 3.5H_2O \rightarrow UO_2F_2.4HF.1.5H_2O$$

For the NCSE calculations, the composition of the breakdown product is simplified to $UO_2F_2.3.5H_2O$ that represents the same H/U of 7 as above.

The determination of the Passive Engineered Control (PEC) values for the major controlling parameters used to control criticality in the facility is described below.

The following criticality analyses are performed assuming optimum moderation (i.e., various H/U ratios greater than and less than 7 are analyzed) and 30 cm water reflection to determine the critical and safe values for various parameters. Although the facility will be limited to 10.0 wt.% enrichment, as additional conservatism, the analysis is also performed at 11.0 wt.% enrichment and the PEC values for the buildings/systems/components are developed based on the analysis at 11.0 wt.% enrichment, for k_{eff} of 0.95, except for the Contingency Dump System upset events (both the dump traps and to tails cylinders) which are limited to 1.5 wt.% ^{235}U for the dumped material.

A uniform solution of UO_2F_2, and a fixed enrichment, 10.0 wt.% and 11.0 wt.% enrichment, are conservatively modeled using SCALE 6.2.4 (ORNL 2020) is and the ENDF/B-VII.1 cross-section library.

Table 5-2 shows the results of the criticality analyses for k_{eff} = 1, critical values, and for k_{eff} = 0.95, safe values.

Table 5-3 lists the safety criteria of Table 5-2 which are used as PEC control parameters to prevent a nuclear criticality event. The product cylinders are only safe under conditions of limited moderation and enrichment. In such cases, both design and operating procedures are used to ensure that these limits are not exceeded. Centrifuge array criticality is precluded by a probability argument with multiple operational procedure barriers.

Table 5-2 Critical and Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2

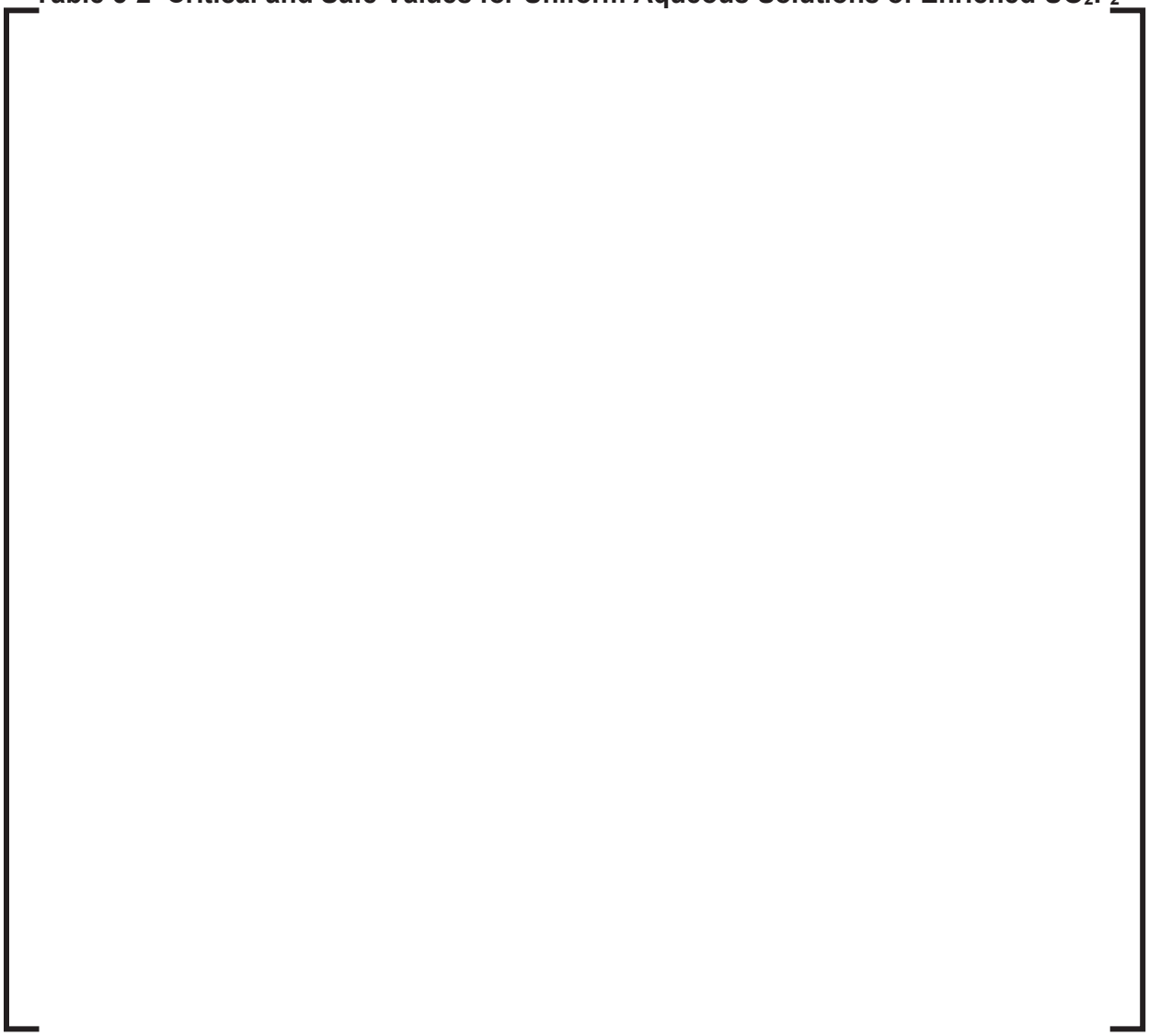

Table 5-3 Safety Criteria for Buildings / Systems / Components

The major controlling parameters used in the Facility are enrichment control, geometry control, moderation control, and/or limitations on the mass as a function of enrichment. In addition, reflection, interaction, and heterogeneous effects are important parameters considered and applied where appropriate in NCSEs. NCSEs are used to identify the significant parameters affected within a particular system. All assumptions relating to process, equipment, material function, and operation, including credible abnormal conditions, are justified, documented, and independently reviewed.

The NCSE captures the criticality calculations performed to determine the critical and safe limits of the Facility as well as the control parameters to prevent a nuclear criticality event.

Based on the criticality analyses, the control parameters applied to the Facility are as follows:

Enrichment

Enrichment is controlled to limit the percentage ^{235}U within any process, vessel, or container to a maximum enrichment of 10 wt.% except for the contingency dump system and for certain non-PEC tanks. The design of the contingency dump system controls enrichment to a limit of 1.5 % ^{235}U. The enrichment may further be restricted in non-PEC tanks to ≤ 1.0 wt.% ^{235}U.

Although the Facility is limited to a maximum enrichment of 10 wt.%, as added conservatism, nuclear criticality safety is analyzed using an enrichment of 11 wt.% ^{235}U.

Geometry/Volume

Geometry / volume control may be used to ensure criticality safety within specific process operations or vessels, and within storage containers.

Geometry control is implemented by limiting equipment dimensions in systems whose criticality safety relies on maintaining specific geometric configurations.

Prior to beginning operations, in response to changes to operations, and at periodic intervals, all dimensions relied on in demonstrating subcriticality are verified. Relevant dimensions and material properties are maintained in the facility's configuration-management program.

Credible mechanisms for loss of geometry control-such as corrosion, leakage, bulging, transfer into unfavorable geometry, or changes to a more reactive physicochemical form-are evaluated, and controls are established as necessary.

Neutron interaction with any equipment containing fissile material is considered in the demonstration of subcriticality unless the criteria for neutronical isolation can be met or justified.

The geometry / volume limits are chosen to ensure $k_{eff} < 0.95$.

The PEC values of geometry / volume define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming 11 % ^{235}U for safety margin.

Volume limits are used as specified in NCSEs. PEC value of geometry / volume is evaluated assuming the most reactive credible geometry (sphere), optimum moderation, and full water reflection.

When volume control is used, the size of the containers is ensured through the Criticality Management Program and/or by procedural requirements specifying the containers permitted for fissile-material operations.

Moderation

Physical structures are designed to preclude the ingress of moderators.

Water and oil are the moderators considered in the Facility. Moderation control for product cylinders is established consistently with the guidelines of ANSI/ANS-8.1-2014 (ANSI 2023a) and incorporates the criteria below:

- Controls are established to limit the amount of moderation entering the cylinders.
- When moderation is the only parameter used for criticality control, the following additional criteria are applied. These controls ensure that at least two independent controls would have to fail before a criticality accident is possible.
    - Two independent controls are utilized to verify cylinder moderator content.
    - These controls are established to monitor and limit uncontrolled moderator prior to returning a cylinder to production thereby limiting the amount of uncontrolled moderator from entering system to an acceptable limit.
    - The evaluation of the cylinders under moderation control includes the establishment of limits for the ratio of maximum moderator-to-fissile material for both normal operating and credible abnormal conditions. This analysis has been supported by parametric studies.
- When moderation is not considered a control parameter, either optimum moderation or worst-case H/U ratio is assumed when performing criticality safety analysis.

Firefighting procedures for use in moderation-controlled areas are evaluated in CSEs. Restrictions on the use of moderating firefighting agents are included in procedures and training. The effects of a fire and activation of fire suppression are evaluated.

The centrifuge process equipment is a closed system designed to treat gaseous UF_6. The closed system minimizes the introduction of moderation.

Mass

Mass control may be utilized to limit the quantity of uranium within specific process operations, vessels, or storage containers. Mass control may be used on its own or in combination with other control methods. Weighing or physical measurements is employed to verify the mass of the material.

Conservative administrative limits for each operation are specified in the operating procedures.

When the dimensions of equipment or containers with a fixed geometry are used to limit the mass of the material, a conservative process density is used to calculate the resulting mass.

Whenever mass control is established for a container, records are maintained for mass transfers into and out of the container. Establishment of mass limits for a container involves consideration of potential moderation, reflection, geometry, spacing, and enrichment. The evaluation considers normal operations and credible abnormal conditions for determination of the operating mass limit for the container and for the definition of subsequent controls necessary to prevent reaching the safety limits. When only administrative controls are used for mass-controlled systems, double batching is conservatively assumed in the analysis.

In the absence of a dedicated control, an item containing enriched uranium is assumed to hold the maximum credible 235-U content based on its available volume.

Reflection

Normal and credible abnormal reflection is considered when performing NCSEs.

The possibility of full water reflection is considered when performing analyses, for example for the determination of PEC values when reflection is not controlled. It is recognized that concrete can be a more efficient reflector than water, and its potential presence is considered.

Minimum reflection conditions equivalent to a 1-inch tight-fitting water reflector are assumed to account for personnel and other transient incidental reflectors not explicitly included with fixed reflectors in the model.

Interspersed moderation is considered when evaluating subcriticality for an array of units.

Reflection controls are used to limit the potential reactivity of a fissile material operation.

Interaction

NCSEs consider the potential effects of interaction, including interaction effects of in-transit materials. Spacing requirements will be determined on a system-by-system basis. Individual unit multiplication, array interaction, and in-transit material interactions are evaluated in the NCSEs.

Passive engineered controls are used to the extent possible to ensure spacing requirements are maintained. When implemented, the structural integrity of spacers, storage racks, etc. is sufficient to ensure subcriticality under normal and credible abnormal conditions. Engineered devices that are moveable (e.g., birdcage drums, 55-gallon drums) are inspected periodically for deformation.

5.7.4 Safety Margins Against Criticality

The Facility UF_6 systems involve mostly gaseous operations. These operations are carried out under reduced atmospheric conditions (vacuum) or at slightly elevated pressures. The following contributes to the safety margins against criticality:

- It is highly unlikely that any size changes of process equipment under these conditions would lead to a criticality situation because a volume or mass limit may be exceeded.
- The Facility design has minimized the possibility of accidental moderation by eliminating direct water contact with accumulated UF_6.
- The Facility's stringent procedural controls for enriching UF_6 ensure that it does not become unacceptably hydrogen moderated while in process. The Facility's UF_6 systems operating procedures contain safeguards against loss of moderation control, ANSI/ANS-8.1-2014 (ANSI 2023a). No neutron poisons are relied upon to assure criticality safety.
- All the equipment and cylinders accumulating product and feed UF_6 are within enclosures protecting them from water ingress.

5.8 ADDITIONAL NCS PROGRAM COMMITMENTS

No additional NCS program commitments are made beyond the commitments already made previously.

5.9 REFERENCES

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NRC. 2018. "Nuclear Criticality Safety Standards for Fuels and Materials Facilities." *U.S. Nuclear Regulatory Commission* Regulatory Guide 3.71 (Revision 3).

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Chapter 6 – Chemical Process Safety

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6.0 CHEMICAL PROCESS SAFETY

This chapter presents the Orano Enrichment USA LLC (OE) strategy for managing chemical process safety in accordance with the regulatory requirements of 10 CFR Part 70. The objective demonstrates that OE's chemical safety controls provide reasonable assurance that the health and safety of the public and facility personnel are protected from chemical hazards associated with licensed operations.

The chapter outlines the following key components of OE's chemical process safety program:

- Chemical Screening and Classification: A systematic approach to identifying and categorizing chemicals based on their physical and health hazards, consistent with federal standards and industry best practices.
- Hazard Consequence Evaluation: Methodologies used to assess the potential impacts of hazardous chemical releases, including airborne dispersion modeling, exposure assessment, and consequence thresholds.
- Chemical Safety Assurance: Features Description of engineered controls, administrative procedures, and safety systems designed to prevent, detect, and mitigate chemical hazards.

The OE chemical process safety program is designed to meet the following regulatory and guidance criteria:

- NUREG-1520, Chapter 6 – NRC Standard Review Plan for evaluating chemical safety in fuel cycle facilities (NRC 2015a)
- 10 CFR 70.61 – Performance requirements for safety
- 10 CFR 70.62 – Safety program implementation
- 10 CFR 70.64 – Requirements for new processes and modifications
- NUREG-1601 – Chemical process safety guidance for NRC licensees (NRC 1997)

6.1 CHEMICAL SCREENING AND CLASSIFICATION

Table 6-1 provides the listing of chemicals that are expected to be in use at the Facility in quantities where they require stored inventory. Chemical formulas in this Chapter utilize subscripting per standard convention. The hazardous classification of each chemical is presented as defined in the International Fire Code (IFC 2021). Although not expressly identified as a hazardous classification in the IFC, a column is provided to identify chemicals that are radioactive.

Each chemical is classified into one of three categories: Chemicals of Concern (Class 1), Interaction Chemicals (Class 2), or Incidental Chemicals (Class 3).

The definition of each classification is provided below.

Table 6-2 through Table 6-6 are the basic chemical inventories for the enrichment-related process structures and support areas at the Facility. Each of these tables lists a major facility structure or area and the associated inventory of significant chemicals stored or used for each area. These tables do not include the listing of all incidental effluents, sludges, wastes and waste streams, and other incidental chemicals characterized as Class 3 materials that may be present. These chemicals are not a process safety concern as they have no ability to impact licensed material systems in a manner affecting 10 CFR 70.61 performance requirements. Inventories of solid

wastes, gaseous and liquid effluents, sludges, and other chemical containing waste streams that will be processed and/or discharged are detailed in Section 4.13 of the Environmental Report.

6.1.1 Chemicals of Concern (Class 1)

Chemicals of Concern (Class 1) are determined based on one or more characteristics of the chemical and/or the quantity in storage/use at the facility. For licensed material or hazardous chemicals produced from licensed materials, chemicals of concern are those that, in the event of release, have the potential to exceed any of the concentrations defined in 10 CFR 70.61 as listed below.

<u>High Risk Chemicals of Concern</u>

1. An acute worker dose of 1 Sv (100 rem) or greater total effective dose equivalent.
2. An acute dose of 0.25 Sv (25 rem) or greater total effective dose equivalent to any individual located outside the controlled area.
3. An intake of 30 mg or greater of uranium in soluble form by any individual located outside the controlled area.
4. An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that:
   (i) Could endanger the life of a worker, or
   (ii) Could lead to irreversible or other serious, long-lasting health effects to any individual located outside the controlled area.

<u>Intermediate Risk Chemicals of Concern</u>

1. An acute worker dose of 0.25 Sv (25 rem) or greater total effective dose equivalent.
2. An acute dose of 0.05 Sv (5 rem) or greater total effective dose equivalent to any individual located outside the controlled area.
3. A 24-hour averaged release of radioactive material outside the restricted area in concentrations exceeding 5,000 times the values in Table 2 of Appendix B to 10 CFR 20.
4. An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that:
   (i) Could lead to irreversible or other serious, long-lasting health effects to a worker, or
   (ii) Could cause mild transient health effects to any individual located outside the controlled area.

<u>Non-Licensed Chemicals of Concern</u>

For those chemicals that are not related to licensed materials, chemicals of concern are those that are listed and handled above threshold quantities of either of the following standards:

1. 29 CFR 1910.119 – OSHA Process Safety Management
2. 40 CFR 68 – EPA Risk Management Program.

These chemicals represent, based on their inherent toxic, reactive, or flammable properties, a potential for large, airborne chemical release and/or acute chemical exposure to an individual that:

   (i) Could endanger the life of a worker, or

(ii) Could lead to irreversible or other serious, long-lasting health effects to any individual located outside the controlled area.

It is noted here that uranium hexafluoride (UF_6) is the only licensed material-related chemical of concern (Class 1) used at the Facility. There are no non-licensed chemicals of concern at the Facility.

6.1.2 Interaction Chemicals (Class 2)

Interaction chemicals (Class 2) are those chemicals/chemical systems that require evaluation for their potential to precipitate or propagate accidents in chemical of concern (Class 1) systems, but by themselves are not chemicals of concern.

6.1.3 Incidental Chemicals (Class 3)

The Facility uses other chemicals that are neither chemicals of concern nor interaction chemicals. Some of these incidental chemicals (Class 3) include those that have the potential to result in injurious occupational and/or environmental exposure but represent no potential for acute exposure to the public and which via their nature, quantity, and/or use, have no potential for impacting chemicals of concern (Class 1).

These chemicals are not subject to chemical process safety controls. Controls placed on incidental chemical storage, use and handling as necessary and as follows:

1. General occupational chemical safety controls are in place for protection of facility employees in the storage, handling, and use of all chemicals as required by 29 CFR 1910
2. Environmental protection controls required to prevent and/or mitigate environmental damage due to spills and discharges and to control anticipated effluents and waste are detailed in Chapter 4.13, “Waste Management” of the Facility Environmental Report.

Table 6-1: Chemical Hazard Classification [Note 1]

Chemical	Class	Formula	Phase(s) Note 2	Radioactive	Toxic	Corrosive	Water Reactive	Flammable	Combustible	Oxidizer	Other	Comments
uranium hexafluoride [Note 3]	1	UF_6	S/L/G	•	•	•	•					
uranic compounds	NA	UO_2F_2, UF_4, U_3O_8	S/L	•	•	•	•					UF_6 reaction byproducts, deposits & in solution
hydrogen fluoride	NA	HF	G		•	•	•					UF_6 reaction byproduct
sodium fluoride	2	NaF	S		•							granules
aluminum oxide (activated)	2	Al_2O_3	S								•	irritant, powder / granules
paper, polymers	3		S						•			ventilation filter media, anti-contamination clothing, ion exchange resin, etc.
potassium hydroxide	3	KOH	S		•	•						
phosphate	3		S								•	surfactant, irritant, P-3 Plastoclin 4100 B
scrap metals	3		S	•								contaminated scrap/parts
citric acid	2	$C_6H_8O_4$	S/L			•						crystals & solution (5-10%)
sodium hydroxide	3	NaOH	S/L		•	•						powder & solution (0.1N)
hydrocarbon oils / greases	3	varies	S/L						•			
hydrocarbon sludges	3	varies	S/L						•			
perfluoropolyether fluids	2	varies	L								•	irritant, long chain perfluorocarbons
methylene chloride	3	CH_2Cl_2	L								•	Health hazard
polydimethylsiloxane (silicone oil)	2	varies	L						•			
hydrocarbon / polar solvents and liquids	3	varies	L					•				ethanol, acetone, toluene, petroleum ether, paint, cutting oils
nitric acid	3	HNO_3	L			•						(50-70%) weight concentration
hydrofluoric acid	3	HF (H_2O)	L			•						38% weight concentration
hydrogen peroxide	3	H_2O_2	L							•		
sulfuric acid	3	H_2SO_4	L			•						
phosphoric acid	3	H_3PO_4	L			•						(10-25%) weight concentration
diesel fuel	3	varies	L						•			generator / vehicle fuel

Chemical	Class	Formula	Phase(s) Note 2	Radioactive	Toxic	Corrosive	Water Reactive	Flammable	Combustible	Oxidizer	Other	Comments
deionized water	3	H_2O	L			•						
hydrofluorocarbons (HFC) - hydrofluoroolefins (HFO)	3	varies	L/G								•	refrigerant, irritant
nitrogen	2	N_2	L/G								•	asphyxiant, test gas / purge gas
propane	3	C_3H_8	L/G					•				test gas
hydrogen	3	H_2	G					•				test gas
acetylene	3	C_2H_2	G					•				welding gas
oxygen	3	O_2	G							•		test gas / welding gas
argon	3	Ar	G								•	asphyxiant, test gas / welding gas
helium	3	He	G								•	asphyxiant, test gas

Notes:

1: Hazardous material classifications per the International Fire Code **(IFC 2021)**. Radioactive classification has also been included although not identified as a specific IFC classification.
2: Lists the phases applicable based on facility use of chemical; S – solid, L – liquid, G – gas/vapor.
3: Solid UF6 cylinders also have ullage space containing vapor UF6 and traces of HF, air, non-condensables and U non-volatiles (<1% total wt)

Table 6-2: Chemical Inventory – Separations Building Module (SBM) [Note 1] and Blending, Sampling and Preparation Building (BSPB)

(Security-Related Information Withheld Under 10 CFR 2.390)

Chemical	Phase	Inventory by Location				Comments
		UF_6 Annex	Module	Process Service Corridor	Blending, Sampling and Preparation Building	
uranium hexafluoride	[ ]	[ ]		[ ]	[ ]	Note 2
uranium hexafluoride	[ ]				[ ]	Note 2
uranium hexafluoride	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
uranic compounds	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
hydrogen fluoride	G	trace	trace	trace	trace	residual byproducts in piping
sodium fluoride	S	580 kg (1,279 lb)		7,820 kg (1.72 E+04 lb)	144 kg (317 lb)	chemical traps
aluminum oxide	S	315 kg (694 lb)		350 kg (772 lb)	172 kg (379 lb)	chemical traps
paper, polymers	S	varies		varies	varies	ventilation filter media, anti-contamination clothing
perfluoropolyether fluids	L	200 L (52.8 gal)		100 L (26.4 gal)	100 L (26.4 gal)	vacuum pump oil
polydimethylsiloxane (silicone oil)	L	600 L (159 gal)			140 L (37 gal)	
trifluoromethane Note 3	L/G	20 kg (44 lb)			6 kg (13.2 lb)	R23 refrigerant
fluoroethane blend Note 3	L/G	150 kg (331 lb)			40 kg (88 lb)	R1234ze
fluoroethane blend Note 3	L/G	600 kg (1323 lb)			160 kg (353 lb)	R449 refrigerant
nitrogen	L/G	piping	piping	piping	piping	vapor inventory <100 kg (221 lb) in piping; for/during purging

Chemical	Phase	Inventory by Location				Comments
		UF_6 Annex	Module	Process Service Corridor	Blending, Sampling and Preparation Building	
hydrocarbon oils / greases	S/L	varies		varies	varies	miscellaneous lubricants, hydraulic fluid

Notes:

1. Quantities are per individual Separations Building Module (SBM) except for Blending, Sampling and Preparation Building (BSPB) which is a separate building supporting multiple SBMs – the Facility has four SBMs. Inventories are maximum possible operational inventory (i.e., all stations and cylinders are full, NaF traps 100% saturated with UF_6, and similar); routine inventories will be less.
2. BSPB estimate based on: Solid – TBD 48Y cylinders (staged/donor stations and ventilated room), TBD 30B cylinders (staged/receiver stations). Liquid – TBD sampling autoclaves (combined 48Y and 30B).
3. All refrigerants are assumed; types and inventories may change.

Table 6-3: Chemical Inventory – Centrifuge Assembly Building

(Security-Related Information Withheld Under 10 CFR 2.390)

Chemical	Phase	Centrifuge Assembly Area	Centrifuge Test Facility	Centrifuge Post Mortem Facility	Comments
uranium hexafluoride	[ ]		[ ]		[ ]
hydrogen fluoride	G		trace		residual byproduct
uranic compounds	[ ]		[ ]		[ ]
aluminum oxide (activated)	S		20 kg (44 lb)		
Sodium fluoride	S		10 kg (22 lb)		
perfluoropolyether fluids	L	20 L (5.3 gal)			
hydrofluoric acid (38%)	L	2 L (0.53 gal)			
nitric acid (50-70%)	L	2 L (0.53 gal)			
deionized water	L		100 L (26.4 gal)		
phosphoric acid (10 – 25%)	L		50 L (13.2 gal)		Dekopur FS 50
ethanol	L	220 L (58 gal)			
phosphate	S	60 kg (132 lb)			P-3 Plastoclin 4,100 B
propane	L/G	40 kg (88 lb)			
nitrogen	G	12,000 m^3 (4.24 E+05 ft^3)	40 m^3 (1,413 ft^3)		
Nitrogen	L		600 L (159 gal)	600 L (159 gal)	
argon	G	400 m^3 (14,130 ft^3)			
helium	G	2,400 m^3 (8.48 E+04 ft^3)			

Chemical	Phase	Centrifuge Assembly Area	Centrifuge Test Facility	Centrifuge Post Mortem Facility	Comments
hydrocarbon oils/greases	L	60 L (15.9 gal)			
paper, polymers	S	varies	varies		disposable clothing, wipes, gloves, etc. Note 1

Notes:

1. There will be inventories of paper, polymer and related filter media and other metallic and non-metallic solid waste. Quantities by area will be managed in accordance with safety analysis limits and/or regulatory requirements as applicable. Throughput of waste material is detailed in Chapter 4.13, “Waste Management” of the Environmental Report.

Table 6-4: Chemical Inventory – Technical Support Building (TSB) and Operation Support Building (OSB)

(Security-Related Information Withheld Under 10 CFR 2.390)

Chemical	Phase	TSB	OSB	Comments
uranium hexafluoride	[ ]	[ ]		
uranium hexafluoride	[ ]	[ ]		
uranic compounds	[ ]	[ ]		[ ]
hydrogen fluoride	G	trace		residual byproduct in components
sodium fluoride	S	200 kg (440 lb)	240 kg (529 lb)	
aluminum oxide (activated)	S	720 kg (1,587 lb)	40 kg (88 lb)	
paper, polymers	S	varies		ventilation, air and vacuum system filter media, ion exchange resin, disposable clothing, wipes, gloves, etc. Note 1
scrap metals	S	varies		pump, valve, piping parts Note 1
perfluoropolyether fluids	L	40 L (10.6 gal)	260 L (68.7 gal)	
methylene chloride	L	533 L (141 gal)	267 L (70 gal)	
HFC/HFO	L	varies	varies	refrigerants
hydrocarbon / polar solvents and liquids	S/L	varies		ethanol, acetone, toluene, petroleum ether, paint, cutting oils Note 2
hydrocarbon oils / greases	S/L	varies		Note 2
hydrocarbon sludges	S/L	varies		Note 2
hydrogen	G	18 m^3 (636 ft^3)		
acetylene	G	12 m^3 (424 ft^3)		
oxygen	G	22 m^3 (777 ft^3)		
nitrogen	G	20 m^3 (706 ft^3)		
nitrogen	L	4 L (1.1 gal)		
propane	G	2 kg (4.4 lb)		
argon	G	380 L (100 gal)		
citric acid	S	18 m^3 (636 ft^3)		
citric acid (5-10%)	L	2,650 L (700 gal)		
sodium hydroxide	S/L	430 L (114 gal)		0.1N solution, KOH is potential substitute
hydrogen peroxide	L	8 L (2.1 gal)		
sulfuric acid	L	20 L (5.3 gal)		

Chemical	Phase	TSB	OSB	Comments
nitric acid (50-70%)	L	52 L (14 gal)		
phosphoric acid	L	88 L (23 gal)		
diesel fuel	L			1,894 l (500 gal) diesel generator (exterior)

Notes:

1. There will be inventories of paper, polymer and related filter media and other metallic and non-metallic solid waste. Quantities by area will be managed in accordance with safety analysis limits and/or regulatory requirements as applicable. Throughput of waste material is detailed in Chapter 4.13 of the Environmental Report.
2. There will be various oils, paints, solvents, and sludges in waste areas, workshop and laboratory spaces. Quantities by area will be managed in accordance with safety analysis limits and/or regulatory requirements (i.e., Occupational Safety and Health Administration (OSHA), National Fire Protection Association (NFPA), etc.) as applicable. Throughput of waste material is detailed in Chapter 4.13 of the Environmental Report.

Table 6-5: Chemical Inventory – Mechanical Services Building (MSB) and Electrical Services Building (ESB)

Chemical	Phase	Mechanical Services Building	Electrical Services Building	Adjacent Exterior Areas	Comments
paper, polymers	S	varies	varies		ventilation filter media, ion exchange resin Note 1
hydrocarbon oils / greases	S/L	varies	varies		Note 2
diesel fuel	L			113,562 L (30,000 gal)	2 tanks at 56,781 l (15,000 gal)
diesel fuel	L			7,571 L (2,000 gal)	adjacent to gasoline and Diesel Refueling Station
gasoline	L			7,571 L (2,000 gal)	adjacent to gasoline and Diesel Refueling Station
nitrogen	L			75,708 L (20,000 gal)	4 tanks at 18,927 l (5,000 gal)

Notes:

1. There will be inventories of paper, polymer and related media. Quantities by area will be managed in accordance with safety analysis limits and/or regulatory requirements as applicable. Throughput of waste material is detailed in Chapter 4.13, “Waste Management” of the Environmental Report.
2. There will be various oils, solvents, and lubricants associated with mechanical equipment in utility spaces. Quantities by area will be managed in accordance with safety analysis limits and/or regulatory requirements (i.e., Occupational Safety and Health Administration (OSHA), National Fire Protection Association (NFPA), etc.) as applicable. Throughput of waste material is detailed in Chapter 4.13, “Waste Management” of the Environmental Report.

Table 6-6: Chemical Inventory – Exterior Areas

(Security-Related Information Withheld Under 10 CFR 2.390)

6.2 CHEMICAL HAZARDS ANALYSIS

6.2.1 Integrated Safety Analysis (ISA)

OE has prepared an ISA as required under 10 CFR 70.62.

The ISA:

- Provides a list of the accident sequences which have the potential to result in radiological and non-radiological releases of chemicals of concern
- Provides reasonable estimates for the likelihood and consequences of each accident identified
- Applies acceptable methods to estimate potential impacts of accidental releases.

The ISA also:

- Identifies adequate engineering and/or administrative controls (IROFS) for each accident sequence of significance
- Satisfies principles of the baseline design criteria and performance requirements in 10 CFR 70.61 by applying defense-in-depth to high-risk chemical release scenarios
- Assures adequate levels of these controls are provided so those IROFS will satisfactorily perform their safety functions.

The ISA demonstrates that the facility and its operations have adequate engineering and/or administrative controls in place to prevent or mitigate high and intermediate consequences from the accident sequences identified and analyzed.

6.2.2 Consequence Analysis Methodology

This section describes the methodology used to determine chemical exposure/dose and radiochemical exposure/dose criteria used to evaluate potential impact to the workers and the public in the event of material release. This section limits itself to the potential effects associated with accidental release conditions.

6.2.2.1 Defining Consequence Severity Categories

The accident sequences identified by the ISA are categorized into one of three consequence categories (high, intermediate, or low) based on their forecast radiological, chemical, and/or environmental impacts. Section 6.1, Chemical Screening and Classification, presented the radiological and chemical consequence severity limits defined by 10 CFR 70.61 for the high and intermediate consequence categories.

To quantify criteria of 10 CFR 70.61 for chemical exposure, standards for each applicable hazardous chemical must be applied to determine exposure that could: (a) endanger the life of a worker; (b) lead to irreversible or other serious long-lasting health effects to an individual; and (c) cause mild transient health effects to an individual. Per NUREG-1520 (NRC 2015a), acceptable exposure standards include the Emergency Response Planning Guidelines (ERPG 2022) established by the American Industrial Hygiene Association and the Acute Exposure Guideline Levels (AEGL 2004) established by the National Advisory Committee for Acute Guideline Levels for Hazardous Substances. The definitions of various ERPG and AEGL levels are contained in Table 6-7.

The exposure severity limits of 10 CFR 70.61 are summarized in Table 6-8. The severity limits defined in this table come from regulation or known reference criteria. Numerical values for applying these severity categories were developed as defined below.

The toxicity of UF_6 is due to its two hydrolysis products, hydrogen fluoride (HF) and uranyl fluoride (UO_2F_2) that are developed in the event of a UF_6 release at the facility. The AEGL values for HF and UF_6 were utilized for evaluation of chemotoxic exposure. Additionally, since UO_2F_2 is a soluble uranium compound, the values presented in NUREG-1391 (NRC 1991) are utilized for evaluating soluble uranium (U) exposure in terms of both chemical toxicity and radiological dose. In general, the chemotoxicity of uranium inhalation/ingestions is of more significance than radiation dose resulting from internal U exposure. The AEGL values for HF are presented in Table 6-9. The AEGL values for UF_6 are presented in Table 6-10. The values from NUREG-1391 (NRC 1991) for soluble uranium are presented in Table 6-11. These values are used to assess the severity of public exposure to soluble uranium for individuals located outside the controlled area boundary. In this approach, the total amount of uranium inhaled is calculated without taking credit for any reduction in uptake, such as the portion that would normally be exhaled or cleared occur through exhalation or normal respiratory clearance mechanisms. This conservative intake estimate is then compared directly to the body burden limits established in NUREG-1391 (NRC, 1991), which represent the amount of uranium retained in the body and available to cause renal injury. By comparing total intake to these retention based thresholds, the methodology ensures that potential health effects are not underestimated and that consequence categorization remains conservative for public receptors.

The uranium intake limits from NUREG-1391 (NRC 1991) are not applied for worker cases because the worker is more conservatively protected by the UF_6 AEGL limits. At a standard respiration rate, the amount of uranium intake that occurs at AEGL limits is lower than NUREG-1391 (NRC 1991) values.

Table 6-12 summarizes the values used to define consequence severities considering both HF and UF_6 exposures as derived from the AEGL (AEGL 2004) and NUREG-1391 (NRC 1991) values. The assumptions associated with the application of these values are listed below.

6.2.2.1.1 Worker Exposure Assumptions

Individual accidents are hypothesized as a release of UF_6 into the room of concern over a period of time while a worker is present. The UF_6 is assumed to instantaneously and uniformly mix within the room free volume, with no leakage or ventilation losses. Under these bounding conditions, the airborne concentration increases continuously throughout the release period until the release stops. These assumptions provide a conservative basis for estimating worker exposure. It is recognized that a worker located in close proximity to the release point may experience localized concentrations higher than those predicted by the uniform-mixing model. Such conditions occur only during specific operational activities, such as cylinder connection or disconnection. For these tasks, established procedures require the operator to wear respiratory protection, thereby mitigating the potential for elevated near-field exposure. Consequently, the uniform-mixing assumption remains appropriate and conservative for evaluating room-average airborne concentrations for accident analysis purposes.

The consequence to the worker is computed two ways: with the UF_6 unchanged in chemical form; and with UF_6 completely reacted with the humidity in the air to form HF and UO_2F_2. The exposure is evaluated for: radiation dose via inhalation of uranium in the form of soluble UO_2F_2; chemical toxicity from the inhaled uranium or UF_6; and chemical toxicity from the inhaled HF. The worker exposure duration is independent of the release duration, and the consequences are computed on the time-averaged concentration.

Any release from UF_6 systems/cylinders at the facility predominantly consists of HF with some potential entrainment of UO_2F_2 particulate. In addition to activating ambient HF detector alarms, an HF release produces a visible cloud and a pungent, irritating odor. The odor threshold for HF is less than 1 ppm and the irritating effects of HF are intolerable at concentrations well below those that could cause permanent injury or which produce escape-impairing symptoms. Employees are trained in proper actions to take in response to a release and it can be confidently predicted that workers will take immediate self-protective action to escape a release area upon detecting any significant HF odor. Accordingly, ten-minute AEGL values are used to evaluate worker exposure durations, which are 10 minutes or less. Actual releases are detected by the worker(s) who can reliably evacuate all areas of concern within the evaluated exposure durations.

Another assumption made in conducting consequence severity analysis is that for releases precipitated by a fire event, only public exposure is considered in determining consequence severity; worker exposures were not considered. The worker is assumed to evacuate the area of concern once the fire is detected by the worker. Fires of sufficient magnitude to cause chemical/radiological release must be of a severity to either have caused failure of a mechanical system/component or involve substantive combustibles containing uranic content. In either case, the space would be untenable for unprotected workers. Sufficient time is available for the worker to reliably detect and evacuate the area of concern prior to release. Fire brigade/fire department members responding to emergencies are required by emergency response procedure (and regulation) to have suitable respiratory and personal protective equipment.

6.2.2.1.2 Public Exposure Assumptions

Potential exposures to members of the public are also evaluated using conservative assumptions for both exposure concentrations and durations. Exposure is evaluated for consequence severity against chemotoxic, radiotoxic, and radiological dose.

Individual accidents are postulated in the same manner as described for the worker case – a release of UF_6 into the room of concern over a period of time with water vapor mixing to form UO_2F_2 and HF. These chemicals escape from the room through the ventilation system and are carried via atmospheric dispersion to the controlled area boundary where exposure to both HF and UO_2F_2 is assumed. UF_6 is assumed to have completely reacted with humidity in the air by the time the material reaches the controlled area boundary, so the UF_6 AEGL values are not applied for the individual at the boundary. The methodology assumes immediate exposure (does not account for the time of chemical transport to the boundary) and the exposure duration to the individual at the controlled area boundary is independent of the release duration. The consequence to the individual at the controlled area boundary is evaluated against radiation dose via inhalation of uranium in the form of soluble uranyl fluoride, chemical toxicity from the inhaled uranium, and chemical toxicity from the inhaled hydrogen fluoride. The consequences are calculated based on the average concentration over the duration of public exposure which is assumed to be 30 minutes. This is consistent with self-protective criteria for UF_6/HF plumes listed in NUREG-1140 (NRC 1988).

6.2.2.1.3 Environmental Exposure Assumptions

10 CFR 70.61 also requires a limit on the amount of material released to the environment irrespective if such a release results in exposure to an individual. The limit is defined as a 24-hour averaged release of radioactive material outside the restricted area in concentrations exceeding 5,000 times the values in Table 2 of Appendix B to 10 CFR 20. This value is only influenced by the amount of uranyl fluoride released (HF is not radioactive) and is further dependent on the enrichment level of the released material. The methodology developed a

correlation to airborne concentrations of uranium at a given enrichment level. At 10 wt% enrichment, the maximum allowable uranium concentration value for a 24-hour average concentration is 1.7 mg/m^3, provided in the Facility ISA Summary.

6.2.2.2 Chemical Release Scenarios

The Facility ISA Summary presents the evaluation level chemical release scenarios based on the criteria applied in the ISA. Information on the criteria for the development of these scenarios is also provided in the Facility ISA Summary.

6.2.2.3 Source Term

The methodologies used to determine source term are those prescribed in NUREG/CR-6410 (NRC 1998) and supporting documents.

6.2.2.3.1 Dispersion Methodology

In estimating the dispersion of chemical releases from the Facility, conservative dispersion methodologies are utilized. Site boundary atmospheric dispersion factors are generated using a computer code based on Regulatory Guide 1.145 (NRC 1982) methodology. The code is executed using five years of meteorological data collected by eleven meteorological towers (Met Towers) operated by the Oak Ridge National Laboratory (ORNL) . These towers range from 2.49 miles to 6.32 miles around the Facility site and give a detailed estimate of meteorological conditions that would exist at the site.

The specific modeling methods utilized follow consistent and conservative methods for source term determination, release fraction, dispersion factors, and meteorological conditions as prescribed in NRC Regulatory Guide 1.145 (NRC 1982).

For releases inside of buildings, conservative leak path fractions are assumed as recommended by NUREG/CR-6410 (DOE 1998) and ventilation on and off cases are evaluated for consideration of volumetric dilution and mixing efficiency prior to release to atmosphere.

6.2.2.3.2 RASCAL Dispersion Methodology

The NRC recognized dispersion methodology is the RASCAL model, which was developed by the NRC and documented in NUREG-1940 (NRC 2015b).

The specific modeling methods utilized follow consistent and conservative methods for source term determination, release fraction, dispersion factors and meteorological conditions as prescribed by the NRC. The Facility may use the RASCAL with validation and verification documentation.

6.2.2.4 Chemical Hazard Evaluation

This section is focused on presenting potential deleterious effects that might occur as a result of chemical release from the facility. As required by 10 CFR 70, the likelihood of these accidental releases fall into either unlikely or highly unlikely categories.

6.2.2.4.1 Potential Effects to Workers/Public

The Facility ISA Summary presents the evaluation level accident scenarios identified in the Integrated Safety Analysis and presents the potential consequence severities to facility workers

or members of the public. All postulated incidents have been determined to present low consequences to the workers/public, or, where determined to have the potential for intermediate or high consequences, are protected with IROFS to values less than the likelihood thresholds required by 10 CFR 70.61.

6.2.2.4.2 Potential Effects to Facility

All postulated incidents are determined to present inherently low consequences to the facility. No individual incident scenarios are identified that propagate additional consequence to the facility process systems or process equipment. The impact of external events on the facility, and their ability to impact process systems or equipment of concern is discussed in the Facility ISA Summary.

Table 6-7: ERPG and AEGL Level Definitions

Emergency Response Planning Guideline (ERPG)		Acute Exposure Guideline Level (AEGL)	
General Definition	Values intended to provide estimates of concentration ranges above which one could be responsibly anticipate observing health effects.	**General Definition**	Threshold exposure limits for the protection of the general public, which are applicable to emergency exposure periods ranging from 10 minutes to 8 hours. It is believed that the recommended exposure levels are applicable to general population including infants and children, and other individuals who may be sensitive and susceptible.
ERPG-1	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing more than mild, transient adverse health effects or without perceiving a clearly defined objectionable odor.	**AEGL-1** (non-disabling)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience notable discomfort, irritation or certain asymptomatic, non-sensory effects. However, the effects are not disabling and are transient and reversible upon cessation of exposure.
ERPG-2	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair an individual's ability to take protective action.	**AEGL-2** (disabling)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting adverse health effects, or an impaired ability to escape.
ERPG-3	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing or developing life-threatening health effects.	**AEGL-3** (lethality)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience life-threatening health effects or death.

Table 6-8: Consequence Severity Categories based on 10 CFR 70.61

Severity of Consequence	Receptor		
	Worker	**Offsite Public**	**Environment**
Category 3 High Consequence	Radiation Dose: >1 Sievert (100 rem) Chemical Dose: >AEGL-3 for UF_6 >AEGL-3 for HF	Radiation Dose: >0.25 Sievert (25 rem) Chemical Dose: >NUREG 1391 for permanent renal damage for Uranium >AEGL-2 for HF	No values specified.
Category 2 Intermediate Consequence	Radiation Dose: >0.25 Sievert (25 rem) Chemical Dose: >AEGL-2 for UF_6 >AEGL-2 for HF	Radiation Dose: >0.05 Sievert (5 rem) Chemical Dose: >NUREG 1391 for transient renal injury for Uranium >AEGL-1 for HF	Radioactive release >5000 times the values in 10 CFR Part 20, Appendix B, Table 2 (24 hour averaged)
Category 1 Low Consequence	Accidents with lower radiological and chemical exposures than those listed above.	Accidents with lower radiological and chemical exposures than those listed above.	Lesser radioactive release than listed above.

Table 6-9: AEGL Values for Hydrogen Fluoride

AEGL (ppm [mg/m^3])					
	10-min	**30-min**	**1-hr**	**4-hr**	**8-hr**
AEGL-1	1.0 [0.8]	1.0 [0.8]	1.0 [0.8	1.0 [0.8]	1.0 [0.8]
AEGL-2	95 [78]	34 [28]	24 [20]	12 [9.8]	12 [9.8]
AEGL-3	170 [139]	62 [51]	44 [36]	22 [18]	22 [18]

Source: (AEGL 2004)

Table 6-10: AEGL values for Uranium Hexafluoride (as soluble U)

AEGL (mg/m^3)					
	10-min	**30-min**	**1-hr**	**4-hr**	**8-hr**
AEGL-1	3.6	3.6	3.6	NR	NR
AEGL-2	28	19	9.6	2.4	1.2
AEGL-3	216	72	36	9	4.5

NR: Not Recommended due to insufficient data

Source: (AEGL 2004)

Table 6-11: Health Effects of Soluble Uranium

Health Effect	**Uranium burden per kg body weight (mg U/kg)**	**Uranium burden (mg) in 70 kg person**	**Uranium Intake (mg) by 70 kg person**
50% lethality	1.63	114	230
Threshold for permanent renal damage	0.3	21	40
Threshold for transient renal injury or effect	0.058	4.06	8.3
No effect	0.03	2.1	4.3

Source: (NRC 1991)

Table 6-12: Definition of Consequence Severity Categories

	Receptor	High Consequence	Intermediate Consequence
Acute Radiological Doses	Worker	>100 rem TEDE[a]	>25 rem TEDE[a]
	Outside Controlled Area	>25 rem TEDE [a]	>5 rem TEDE [a]
Acute Chemical Exposure	Worker (10-min exposure)	>216 mg UF_6/m^3 [b] (>147 mg U $/m^3$) >139 mg HF/m^3 [c]	>28 mg UF_6/m^3 [b] (>19 mg U/m^3) >78 mg HF/m^3 [c]
	Outside Controlled Area (30-min exposure)	>21 mg U intake [d] >28 mg HF/m^3 [c]	>4.06 mg U intake [d] >0.8 mg HF/m^3 [c]
Radiological Release	Outside Restricted Area (environment)	Not a 10 CFR 70.61 performance requirement	>1.7 mg U/m^3 (24-hr average)

Notes:

[a] TEDE: Total Effective Dose Equivalent
[b] from Table 6-9
[c] from Table 6-10
[d] from Table 6-11

6.3 CHEMICAL SAFETY ASSURANCE

The facility is designed, constructed, and operated such that injurious chemical release events are prevented. Chemical process safety at the facility is assured by designing the structures, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and during any credible accident or external event.

6.3.1 Management Structure and Concepts

The criteria used for chemical process safety encompasses principles stated in NUREG-1601 (NRC 1997). It is also supported by concepts advocated in 29 CFR 1910.119 and 40 CFR, 68, although it is noted here that there are no chemicals at this facility which exceed threshold planning quantities of either standard.

The intent of chemical safety management principles is to identify, evaluate, and control the risk of chemical release through engineered, administrative, and related safeguards.

The chemical safety philosophy for the facility is to apply sufficient control to identify, evaluate, and control the risk of accidental chemical releases associated with licensed material production to acceptable levels in accordance with 10 CFR 70.61(b) and (c).

The identification and evaluation of chemical release risk is developed through the conduct of an ISA. The development of these scenarios, and the dispersion analysis and chemical/radiological dose assessment associated with each accident sequence is performed and conducted in accordance with NUREG/CR-6410 (NRC 1998) as is described in Section 0.

The control of chemical release risk is ensured through numerous features that are described in the following sections.

6.3.2 System Design

The design of chemical process systems includes numerous controls for maintaining safe conditions during process operations. This is accomplished through several means including managing the arrangement and size of material containers and processes, selection and use of materials compatible with process chemicals, providing inherently safer operating conditions (e.g., vacuum handling), providing process interlocks, controls, and alarming within the chemical processes. All of these plant and equipment features help assure prevention of chemical release. Process piping and components, (e.g., centrifuges, traps, vents, etc.) are maintained safe by limits placed on their operating parameters.

With respect to chemical process safety design features recommended in NUREG-1601 (NRC 1997), this section briefly details the features provided for the UF_6 system which is the only chemical of concern (Class 1) process system.

6.3.2.1 Physical Barriers

Double-Walled Piping and Tanks - The UF_6 system piping operates at subatmospheric pressure throughout the plant except for the liquid sampling operation which is conducted within a secondary containment autoclave. As such, UF_6 system piping is not double-walled. Criticality design has been addressed for this vessel.

Liquid Confinement Dikes – Dikes are provided in areas where uranic material is present in solution in tankage. Criticality design constraints were applied to these containment areas. Confinement dikes are also present for chemical spillage control in TSB areas.

Enclosures- Local exhaust decontamination enclosures are utilized for a small number of decontamination operations (e.g., sample bottles, flex hoses). They are not needed for other operations as the levels of specific activity are low. To confine potential HF/uranic material effluent, flexible exhaust hoses connected to the GEVS are provided for locations where UF_6 systems will be opened (e.g., hose connect/disconnect, maintenance, etc.) to capture any fumes remaining after purging operations. GEVS flexible exhaust hoses and fume hoods are present in the TSB where uranic material containers are opened during laboratory and waste handling operations.

Splash Shields – There are no areas where bulk liquid hazardous chemicals will be handled. Lab operations with hazardous chemicals will be conducted in hoods and/or with appropriate personnel protective equipment for these small-scale operations.

Fire Walls – Fire walls are provided to separate UF_6 and uranic material handling areas from other areas of the facility.

Protective Cages – Protective barriers are provided to protect UF_6 system susceptible components (e.g., piping, small equipment) in areas where there is major traffic.

Backflow Preventers and Siphon Breaks – Liquid systems with high uranic content (i.e., not trace waste streams) are provided with means to prevent backflow or siphon. For the UF_6 gaseous piping, design features are provided to prevent UF_6 migration into the few systems which are required to be interconnected to UF_6.

Overflow vessel – UF_6 is not handled in liquid form in any continuous process and any batch handling is performed in small lab quantities or in a secondary containment autoclave. For those systems where uranic material is in solution, overflow protection features are provided.

Chemical Traps and Filters - Chemical traps and filters are provided on vent and ventilation systems which capture UF_6 to remove HF and uranic contaminants prior to any discharge to atmosphere.

6.3.2.2 Mitigative Features

Driving Force Controls – Driving force controls are provided to isolate heating/cooling equipment at UF_6 take-off stations and cold traps as well as other uranic material containing systems. Other driving force controls include relief valves and cut-offs on the nitrogen system to protect the UF_6 system from overpressure.

Solenoid and Control Valves – These types of valves are provided to stop and/or regulate the flow of UF_6 in the event of abnormal operating conditions.

Spray Systems – Spray systems are not provided for vapor mitigation of UF_6 systems or system areas due to criticality control requirements. Fire sprinkler systems are provided in select process areas as described in Chapter 7.

Alarm Systems – Alarm systems are provided which will alarm in the Control Room for abnormal process parameter (e.g., flow, temperature, pressure, level, etc.) conditions in the UF_6 system and some supporting systems. Leak detection is provided through a network of systems designed to identify any release of UF_6 by detection of the hydrolysis product, HF, within the facility. Ambient HF detectors are distributed throughout key process areas, with additional HF monitoring installed at cylinder-filling stations, autoclaves, and within ventilation and exhaust systems. Depending on the required detection threshold and the specific application, the facility uses either electrochemical sensors or laser-based technologies to ensure rapid and reliable detection of HF release.

Alarm measures are in place to notify facility employees of the need to evacuate process areas and/or the facility in the event of a serious chemical release.

6.3.2.3 Baseline Design Criteria and Defense in Depth

The ISA demonstrates that the design and construction complies with the Baseline Design Criteria (BDC) of 10 CFR 70.64(a) and the defense-in-depth requirements of 10 CFR 70.64(b). The design provides for adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material. The Facility is not proposing any facility-specific or process-specific relaxations or additions to applicable BDC features.

6.3.3 Configuration Management

Configuration management includes those controls which ensure that the facility design basis is thoroughly documented and maintained, and that changes to the design basis are controlled. This includes the following:

A. That management commitment and staffing is appropriate to ensure configuration management is maintained
B. That proper quality assurance is in place for design control, document control, and records management
C. That all structures, systems, and components, including IROFS, are under appropriate configuration management.

A more detailed description of the configuration management system can be found in Section 11.1.

6.3.4 Maintenance

The Facility helps maintain chemical process safety through the implementation of administrative controls that ensure that process system integrity is maintained and that IROFS and other engineered controls are available and operate reliably. These controls include planned and scheduled maintenance of equipment and controls so that design features will function when required. Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance function is closely coupled to operations. The maintenance function plans, schedules, tracks, and maintains records for maintenance activities.

A more detailed description of the maintenance program and maintenance management system can be found in Section 11.2.

6.3.5 Training

Training in chemical process safety is provided to individuals who handle licensed materials and other chemicals at the facility. The training program is developed and implemented with input from the chemical safety staff, training staff, and management. A detailed description of the training program can be found in Section 11.3.

6.3.6 Procedures

A key element of chemical process safety is the development and implementation of procedures that help ensure reliable and safe operation of chemical process systems.

Generally, four types of plant procedures are used to control activities: operating procedures, administrative procedures, maintenance procedures, and emergency procedures.

Operating procedures, developed for workstation and Control Room operators, are used to directly control process operations.

A more detailed description of the procedural development and management program can be found in Section 11.4.

6.3.7 Chemical Safety Audits

Audits are conducted to determine that plant operations are performed in compliance with regulatory requirements, license conditions, and written procedures. As a minimum, they assess activities related to radiation protection, criticality safety control, hazardous chemical safety, fire protection, and environmental protection. The Facility chemical process safety functions and areas will be audited at least triennially.

A more detailed description of the audit program can be found in Section 11.5.

6.3.8 Emergency Planning

The Facility Emergency Plan and program includes response to mitigate the potential impact of any process chemical release including requirements for notification and reporting of accidental chemical releases.

Additional information on emergency response can be found in Chapter 7 and in the Facility Emergency Plan.

6.3.9 Incident Investigation and Corrective Actions

A facility wide incident investigation process exists that includes chemical process related incidents. This process is available for use by any person at the facility for reporting abnormal events and potentially unsafe conditions or activities. Abnormal events that potentially threaten or lessen the effectiveness of health, safety or environmental protection are identified and reported to and investigated by the Environmental Health, Safety & Licensing Manager.

A more detailed description of the incident investigation program can be found in Section 11.6.

6.4 REFERENCES

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Chapter 7 – Fire Safety

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7.0 FIRE SAFETY

This chapter documents the fire safety program for the Orano Enrichment USA LLC (OE) enrichment facility (Facility). The fire safety program is part of the overall facility safety program and is intended to reduce the risk of fires and explosions at the Facility. The facility safety program is described in Chapter 3. The fire safety program documents how the Facility ensures fire safety.

The Facility fire safety program meets the acceptance criteria in Chapter 7 of NUREG-1520 (NRC, 2015) and is developed, implemented and maintained in accordance with the requirements of 10 CFR 70.62(a), 10 CFR 70.22, and 10 CFR 70.65. In addition, the fire safety program complies with 10 CFR 70.61, 10 CFR 70.62, and 10 CFR 70.64. NUREG/CR-6410 (NRC, 1998), NUREG-1513 (NRC, 2001), NRC Generic Letter 95-01 (NRC, 1995), NFPA 1 (NFPA, 2024a), and NFPA 801 (NFPA, 2020b) were utilized as guidance in developing this chapter. Summaries of the changes and updates to several of these codes are listed in the Tables at the end of this Chapter.

The basis for providing automatic sprinkler protection in process areas is to meet International Building Code requirements (ICC, 2021) and to follow recommendations of NFPA 801 (NFPA, 2020b) to use sprinklers as the preferred type of automatic fire system – to the extent such use is consistent with criticality safety limits. Automatic sprinkler protection reduces the need to rely on fire brigade (FB) and fire department response, with provisions for non-sprinkler areas that will be covered under Nuclear Criticality Safety (NCS).

7.1 FIRE SAFETY CONTROLS AND MANAGEMENT MEASURES

Fire safety management measures establish fire protection policies for the site. The objectives of the fire safety program are to prevent fires from starting and to detect, control, and extinguish those fires that do occur. The fire protection organization and fire protection systems described in this chapter provide protection against fires and explosions at the Facility.

Fire safety controls are categorized as:

- Eliminate the Hazard
  (e.g., Process configuration or system design makes the likelihood of the accident sequence *not credible*)
- Passive Engineered Controls
  (e.g., fire-rated barriers, door, windows, coatings or other fire rated passive controls for designated areas in the Fire Hazards Analysis),
- Active Engineered Controls
  (e.g., automatic fire alarms for evacuation, hydrants or hose and reel stations, and suppression systems located in buildings and/or areas containing licensed material-at-risk, MAR)
- Administrative Controls
  (e.g., controls for transient combustibles, fuel volumes, LO/TO, Confined Space, or hot work permitting),

Controls are required for some accident sequences in order to meet an acceptable level of risk, based on event likelihoods and dose consequences of 10 CFR 70.61, *Performance Requirements*.

The safety aspects at the programmatic level are governed by the following:

- The Fire Safety Management Program (SMP), along with employee training on this program, will administratively limit the allowable quantity of transient combustibles in in areas containing radiological and/or chemical hazards.
- Fire barriers shall be designed with adequate safety margins such that the total combustible loading allowed to expose the barrier will not exceed the hourly fire resistance rating of the barrier. Typical fire rating periods can be 1 to 4 hours, and disseminate from the testing standard requirements. Fire rated materials typically will be within the scope of the following examples of consensus standards that are provided here only as guidance:
  - NFPA 252, *Standard Methods of Fire Tests of Door Assemblies* (NFPA, 2022c)
  - NFPA 257, *Standard on Fire Test for Window and Glass Block Assemblies* (NFPA, 2022d)
  - ASTM E119*, Standard Test Methods for Fire Tests of Building Construction and Materials* (ASTM, 2026); supersedes NFPA 255 and NFPA 256
  - ANSI/UL 263, Standard for Safety of Fire Tests of Building Construction Materials (UL, 2011); supersedes NFPA 255 and NFPA 256
  - NFPA 1, *Fire Code*, Section 12.7.6 addresses opening protectives in fire-rated assemblies (NFPA, 2024a)
- UF_6 cylinders contain a majority of the MAR onsite, and are certified with fire rating of 800°C for 30 minutes, in conjunction with certifications that disseminate from the following consensus standards:
  - ANSI 14.1, Uranium Hexafluoride Packagings Transport (ANSI, 2023)
  - ISO 7195, Packagings for the transport of uranium hexafluoride (UF6) (ISO, 2020)
- Pre-action fire sprinkler systems are designed for protected areas onsite by hazard classes following guidance from NFPA 1 (NFPA, 2024a), NFPA 101 (NFPA, 2021b), and some of the following considerations based on the type of fire system:
  - Fire protection water supply including fire water storage tanks, underground mains, hydrant, and water distribution systems following guidance of NFPA 22 (NFPA, 2023b);
  - Fire sprinkler piping, sprinklers, and other components following guidance of NFPA 13 (NFPA, 2025d);
  - Circuitry, wiring, raceways, electronic hardware and software, and primary and secondary power supplies for all systems and sub-components;
  - Primary power supply boundaries ending at the first upstream supply breaker from the fire protection component being supplied (i.e., electric fire pump, local fire alarm panels, fire pump controller) following guidance of NFPA 110 (NFPA, 2025a);
  - Area-wide smoke and/or fire detectors, manually actuated alarm pull boxes, and fire alarm control panels following guidance of NFPA 72 (NFPA, 2019b);
  - Emergency lighting with emergency backup power supplies following guidance of NFPA 1 (NFPA, 2024a) and NFPA 101 (NFPA, 2021b).
- Fire protection systems will be designed to a performance level equivalent to the Natural Phenomena Hazards (Refer to ISAS §1.1.5) required by the International Building Code (ICC,

2021); which will be the effective code of record when said features are submitted to the City of Oak Ridge, TN for construction permitting and inspection.

- Fire protection measures for fuel dispensing areas will follow guidance of NFPA 30A (NFPA, 2024c) and/or NFPA 329 (NFPA, 2025b), with consequences (e.g., fuel fires, deflagrations, etc.) analyzed based on methodologies within the EPA Risk Management Program Guidance (EPA, 2009)
- Laboratory areas (refer to ISAS §2.2.17, etc.) with chemicals will follow guidance of NFPA 45 (NFPA, 2024e) at the Facility where applicable;
- Quality Assurance program requirements provide assurance that fire protection systems are designed, fabricated, erected, tested, maintained, and operated so that they will function as intended based on applicable testing standards.
- Administrative controls will be disseminated from the Fire Safety Management Programs (SMP) requirements as management measures for building evacuation plans, fire drills, and interfacing with the local emergency management and fire department.

7.1.1 Management Policy and Direction

OE is committed to ensuring that the IROFS identified in the ISA Summary are available, reliable, and that the Facility maintains fire safety awareness among employees, maintains control of transient ignition sources, combustibles, and maintains a readiness to extinguish or limit the consequences of fire. The Facility maintains fire safety awareness among employees through its SMPs through the General Employee Training Program. The programs associated with management measures are described in Chapter 11 of this license application (LA).

The responsibility for fire protection rests with the Environmental Health, Safety & Licensing (EHS&L) Manager. The personnel qualification requirements for the EHS&L Manager and information on the OE chain-of-command are presented in LA Chapter 2.

Responsibility for fire protection encompasses the following:

- Fire protection program and procedural requirements
- Fire safety considerations
- Maintenance, surveillance, and quality of the Facility fire protection features
- Control of design changes as they relate to fire protection
- Documentation and record keeping as they relate to fire protection
- Fire prevention activities (e.g., SMPs, management measures, their credited administrative controls, and employee training)
- Organization and training of the FB
- Pre-fire planning and drills

Changes to the Fire Safety Program (FSP) are controlled by the Configuration Management SMP, which is discussed in LA Chapter 11, based on the requirements of 10 CFR 70.72.

7.1.2 Fire Prevention

Management measures will utilize administrative controls with respect to fire safety to maintain the performance of the fire protection systems and delineate the responsibilities of on-site

personnel. The primary fire safety administrative controls are those that relate to fire prevention. These fire prevention controls, in the form of procedures, primarily control the storage and use of combustible materials and the use of ignition sources. These controls include, but are not limited to, the following:

- Governing the handling of transient combustibles in buildings containing IROFS, including work-generated combustibles
- Implementing a (e.g., “Hot Work”) permit system to control ignition sources that may be introduced by welding, flame cutting, brazing, or soldering operations following guidance from NFPA 51B (NFPA, 2024f).
- Ensuring that the use of open flames or combustion-generated smoke for leak testing is not permitted within the Enrichment Facility.
- Conducting routine fire prevention inspections to (1) ensure that transient combustibles adhere to established limits based on the Fire Hazard Analysis; (2) ensure the availability and acceptable condition of fire protection systems/equipment, fire stops, penetration seals, and fire-retardant coatings; and (3) ensure that prompt and effective corrective actions are taken to correct conditions adverse to fire protection and preclude their recurrence
- Performing routine housekeeping inspections of the facility, with a focus on areas having special nuclear material (SNM), referred to as material at risk (MAR) in Appendix A of this FHA.
- Implementing a permit system to control the disarming of fire detection or fire suppression systems, including appropriate compensatory measures (e.g., Fire Watch)
- Implementing fire protection system inspection, testing, and maintenance procedures.

7.1.3 Inspection, Testing and Maintenance of Fire Protection Systems

An inspection, testing and maintenance program is implemented to ensure that fire protection systems and equipment remain operable and function properly when needed to detect and suppress fire following guidance of NFPA 25 for water-based fire systems (NFPA, 2023a), NFPA 70B for maintaining related electrical systems (NFPA, 2026).

Fire protection procedures are used to address such topics as Inspection, Testing and Maintenance training of the onsite Fire Brigade (FB), reporting of fires, emergency response, and for other various management measures. The Facility's Safety, Security, and Emergency Preparedness Manager has responsibility for fire protection procedures; with the Facility's maintenance section having responsibility for the preparation and upkeep of fire protection procedures such as control of repairs to facility penetration seals. Refer to Chapter 11 for additional information on procedures and maintenance activities.

7.1.4 Emergency Response Organization, Qualifications, Drills and Training

The qualifications, drills and training of the FB members who are part of the onsite Emergency Response Organization following guidance of 29 CFR 1910.156 and NFPA 600 (NFPA, 2025e). The primary purpose of the FB Training Program is to develop a group of Facility employees trained in fire prevention, firefighting techniques, first aid procedures, and emergency response. They are trained and equipped to function as a team for fighting fires.

The FB Training Program provides entrance and educational requirements for FB candidates as well as the medical- and job-related physical requirements. The FB Training Program provides

for initial training of all new FB members, semi-annual classroom training and drills, annual practical training, and leadership training for FB Leads.

Refer to Chapter 8 for discussion on the Facility Emergency Management (EM) Plan and the use of off-site emergency response organizations, drills and training. Routine drills will also be performed with off-site emergency response personnel to ensure their capabilities during an emergency event at the OE Facility.

7.1.5 Pre-Fire Plans

Detailed pre-fire plans will be developed for use by the Facility's FB following guidance of FHA, its disseminating protocols, 29 CFR 1910.156 and NFPA 600 (NFPA, 2025e).

The pre-fire plans include the site location, chemical and radiological hazard areas, fire protection equipment (e.g., fire hydrants, site-specific PPE, etc.); approach paths for emergency response, potential explosive hazards in the area (e.g., fuel storage, pressure vessels, etc.); site-evacuation and -accountability protocols for search and rescue; means of isolating electrical, ventilation, and other plant support systems, identification and protection of critical plant equipment, and other information deemed necessary by the local, off-site Fire Department and Emergency Response personnel.

7.2 FIRE HAZARDS ANALYSIS

A Fire Hazards Analysis (FHA) has been conducted evaluating fires at the Facility which, if uncontrolled, could cause a release of Uranium Hexafluoride (UF_6) in quantity and form that may result in an intermediate or high consequence, as defined in 10 CFR 70.61. UF_6 is present in sufficient quantity for these consequence levels to occur in the following areas:

- Separations Building Modules (SBM),
- Cylinder Receipt and Shipping Building (CRSB),
- UF_6 Annexes, Technical Support Building (TSB),
- Product Blending,
- Sampling and Preparation Building (BSPB), and
- Cylinder Storage Pads.

The FHA develops credible, conservatively bound fire scenarios and then assesses the consequences of unmitigated fire events that consist of the following:

- A description of the facility's use and function
- The boundaries of fire areas
- The specific fire hazards and potential fire scenarios within the fire areas
- The methods of consequence analysis
- The occupancy and construction requirements
- Life safety requirements
- Methodology for evaluating the impact of fire on IROFS
- The Facility response to fires
- Defense and mitigation strategy for overall facility protection

The results of the FHA are utilized in the ISA to identify potential fire initiators and accident sequences leading to radiological or toxic chemical consequences, particularly the interaction of fire-initiating events with UF_6 material or its byproducts.

The FHA is updated and controlled through the Configuration Management Program from Chapter 11 based on the requirements of 10 CFR 70.72. This management measure ensures that the information and analyses presented in the FHA are consistent and up-to-date with the current state of the Facility. The FHA is reviewed and updated as necessary to incorporate significant changes and modifications to the Facility, its processes, or combustible inventories.

7.3 FACILITY DESIGN

The design of the Facility incorporates the following:

- Limits on areas and equipment subject to contamination
- Design of structures, equipment, and utilities to facilitate decontamination.
- Design to mitigate and prevent Natural Phenomena Hazard and External hazard events.

7.3.1 Building Construction

Chapter 1 of the ISA describes the building construction, functional areas, and process-related buildings that make up the Facility.

7.3.2 Fire Area Determination and Fire Barriers

The Facility is subdivided into fire areas by barriers with fire resistance ratings for specific hazards, as required by the IBC (ICC, 2021) or by the FHA to ensure licensed material safety consistent with the ISA Summary of Chapter 3. The design and construction of fire barrier walls following guidance of NFPA 221 (NFPA, 2021a) and NFPA 101 (NFPA, 2021b). These fire areas are provided to limit the spread of fire, protect personnel and limit the consequential damage to the Facility. The fire resistance rating of fire barrier assemblies are fire-rated following guidance of ANSI/UL 263 (UL, 2011)and ASTM E119 (ASTM, 2026). Openings in fire barriers are protected consistently with the designated fire resistance rating of the barrier. Door openings in fire rated barriers are protected with fire rated doors, frames and hardware following guidance of NFPA 80 (NFPA, 2016).

7.3.3 Electrical Installation

All electrical systems at the Facility are following guidance of NFPA 70 (NFPA, 2020d).

The Electrical System will be designed to minimize the combustible content of the equipment and the wiring. IROFS are designed to be fail-safe for total loss of electrical power supply. The following systems are monitored:

- 161 kV – 13.2 kV Electrical Switchyard
- 13.2 kV Switchgear
- 480/440 V Switchgear
- Standby Diesel Generators (NFPA 110)
- UPS Systems (NFPA 111)

7.3.4 Electrical Distribution

7.3.4.1 Utility Feed

Power Lines

Power is provided by four 161 kV transmission lines from the existing substations on the Tennessee Valley Authority (TVA) grid. Any of two lines are related to supply the total power requirements of the Facility.

Transformers

In the TVA/Oak Ridge switchyard, two transformers convert 161 kV to 13.2 kV. Each transformer is rated to supply 100% of the total power requirements of the Facility. The power requirements of the Facility is approximately 40 MW.

Distribution into Plan:

Oak Ridge switchyard transformers supply two 13.2 kV trains. Each train can be powered by each Oak Ridge transformer. Redundant 13.2 kV train connect to two switchgears. Distribution system transformers step down 13.2 kV to facility-required voltages (480 V, 120 V, etc.).

7.3.4.2 Plant Distribution

Normal Power:

All normal equipment loads are fed from electrical distribution equipment which is not backed-up by either a Standby Diesel Generators or UPS. The normal load power distribution equipment is served by the following:

- Redundant 161 kV lines
- Two main transformers
- Redundant 13.2 kV switchgears

Short-Break Power (Generator Backup):

Standby Diesel Generators are provided to power equipment that can tolerate a short break in the normal power supply, which allows an orderly shutdown of the Facility. Each of the Standby Diesel Generators is sized for 100% of the short break load of the equipment to which it is connected. The Standby Diesel Generators are not required for safety operation of the Facility and are installed to provide protection of investment.

No-Break Power (UPS):

Uninterruptible Power Supply (UPS) systems are provided to power the facility process equipment that does not tolerate a break (no break load) in the normal power supply. Batteries power the UPS if all other input power is lost. Each of the UPS systems is sized for 100% of its connected load.

Independent No-Break Power (UPS) Loads:

Additional UPS systems with battery backup are installed to provide no break power to support systems such as emergency lighting. These systems are sized and located as necessary to meet the requirements of the equipment served.

7.3.5 Life Safety

The buildings are provided with means of egress, illumination, and protection in accordance with the IBC (ICC, 2021), and following guidance of NFPA 101 (NFPA, 2021b). Barriers with fire resistance ratings consistent with IBC (ICC, 2021), and following guidance of NFPA 101 (NFPA, 2021b), NFPA 221 (NFPA, 2021a) and the FHA are provided to prevent unacceptable fire propagation that could impact personnel egress.

Marking of means of egress, including illuminated exit signs, following guidance of NFPA 101 (NFPA, 2021b) and Chapter 10 of the IBC (ICC, 2021).

Fire Safety and Emergency signage and symbols for egress following guidance of NFPA 170, Standard for Fire Safety and Emergency Symbols (NFPA, 2024b). Routine and emergency Confined Space Entry operations are performed following guidance of NFPA 350 (NFPA, 2022a).

The Physical Security Plan (PSP) addresses the establishment of permanent and temporary Controlled Areas and identifies the ingress and egress methodology during both normal and emergency conditions. These conditions also include access for both onsite and offsite emergency response personnel. Two means of access to the site are provided via (1) one of two controlled gates continuously manned by OE Facility Security and (2) designated emergency access gates (i.e., crash gates). Refer to the OE Facility's PSP for additional details.

7.3.6 Ventilation

The various facility ventilation systems are described in ISAS §2.1.9 and §2.2.16.

The system is designed to perform the required function for exhaust handling of hazardous airborne materials during operations, following guidance of NFPA 91 (NFPA, 2020c). Areas with radiological and chemical hazards for continuous, uninterrupted filtration following guidance of ASME-AG-1, *Code on Nuclear Air and Gas Treatment* (ASME, 2023). Filter train effluents are pre-treated using NaF and Alumina to absorb additional concentrations of gaseous radiological and chemical effluents before reaching these HEPA filter banks.

The ventilation systems installed at the OE Facility are not engineered to exhaust smoke and combustible gases during an onsite, external, or internal facility fire event. This is due to the exhaust streams posing a greater hazard due to the potential of gaseous or particle laden effluents of hazardous materials being further dispersed by transport through ventilation to reach additional onsite and public receptors.

The FHA credits Fire Barriers as IROFS with openings and penetrations to consist of fire-rated doors, fusible linked dampers, and other components to prevent the dispersion of hazardous materials during fire events This design approach to isolate fire events within areas of the Facility also prevents fire events from damaging filter trains and the release of additional hazardous materials "hold up" retained within the filters. All while providing means of egress from an area for onsite personnel during a fire event following guidance of NFPA 101 (NFPA, 2021b), that has superseded NFPA 101B.

7.3.7 Drainage

Water that may discharge from the firewater supply or suppression systems or from firefighting activities could be contaminated with radioactive materials. Discharged water will be contained, stored, sampled, and treated if necessary (i.e., if the water is discharged from an area containing radiological materials). The drainage and containment system design and configuration will be in accordance with the discharge permits provided by the Tennessee Department of Environmental Conservation (TDEC).

Chapter 6 of the Environmental Report (ER) provides detailed criteria for liquid effluent handling practices for the Facility and its surrounding Site (OE, 2026). Requirements for the liquid effluent sampling detection limits are summarized in Chapter 9, Section 9.2. NUREG-1302 (NRC, 1991) and Regulatory Guide 4.16 (NRC, 2010) were followed in determining sample locations, analyses, frequencies, durations, and lower limits of detection.

Chapter 6 of the ER also discusses the Radiological Environmental Monitoring Program (REMP), which supports the Facility's regulatory sampling requirements and discharge of effluents (OE, 2026). Section 9.2 of Chapter 9 has a summary of the REMP sampling performed for environmental monitoring in Table 9-3. The Site's "*Basins*" and the City of Oak Ridge's "*Domestic Sanitary Sewage Treatment Plant*" sampling criteria are provided and are applicable to sampling for firewater runoff and collection.

After sampling, the water from firefighting activities must be dispositioned. Sampling onsite water collection basins will typically be for rainwater collection that will be sampled based on the TDEC permit to discharge the rainwater to natural waterways through an outfall, or equivalent. However, sampling of (previously potable) wastewater from fire events that collects within basins or internal collection tanks and areas (for process water) may not meet the criteria to discharge within the TDEC Permit. For disposition in the City of Oak Ridge's Wastewater Treatment Plant sample results must ensure the wastewater will meet the City's waste acceptance criteria.

ISAS §2.2.11 discusses the disposal options for potentially contaminated liquid effluents at the Facility (that do not meet the discharge criteria of the TDEC Permit for local waterways), per the following two options:

(1) Liquid effluents can be sampled and then sent to the City of Oak Ridge's Water Treatment Plant. ISAS §2.2.6 states that the Facility will be connected to the City of Oak Ridge's wastewater treatment plant for disposition of uncontaminated wastewaters after sampling is performed to ensure constituent levels are within the waste acceptance criteria of the municipal treatment plant.

(2) Contaminated liquid effluents are routed to the onsite Liquid Effluent Collection Treatment System, which is described in more detail within ISAS §2.2.11.

7.3.8 Lightning Protection and External Fire Events

Lightning protection for the OE Facility will be installed as necessary, following guidance of NFPA 780 (NFPA, 2020a). Lightning protection systems are not designed to protect from lightning strikes that may produce external wildfires. NFPA 780 provides guidance for lightning protection systems for safeguarding personnel and property from hazards arising from exposure to lightning. The scope is limited to covering traditional lightning protection systems that are installed on:

- Ordinary structures
- Miscellaneous structures and special occupancies
- Heavy-duty stacks

- Structures containing flammable vapors, flammable gases, or liquids that can give off flammable vapors
- Structures housing explosive materials
- Wind turbines
- Watercraft
- Airfield lighting circuits
- Solar arrays

For external fires from burning debris, campfires, incendiarism, smoking, automobile accidents, lightning or other external causes will be evaluated following guidance of NPFA 1140, *Standard for Wildland Fire Protection* (NFPA, 2022e). External fires that occur near facility structures and buildings following guidance of NFPA 80A, *Recommended Practice for Protection of Buildings from Exterior Fire Exposures* (NFPA, 2022b).

The site-specific FHA evaluates accident sequences for external fire events in §A3.3. The closest hazards are cylinder storage areas that are 30 meters from the site boundary fence. Cylinders are rated for an 800ºC, 30-minute engulfing fire that is not credible to be exceeded at this distance. NUREG-6410 (Appendix C) cites the *EPA Risk Management Program Guidance for Offsite Consequence Analysis*, which covers the methodology for analyzing fuel fires and explosions in both Chapters 5 and 9 (EPA, 2009).

7.4 PROCESS FIRE SAFETY

As stated in Chapter 6, the primary process chemical of concern for the OE Facility is UF_6. UF_6 is not flammable and does not disassociate to flammable constituents under conditions at which it will be handled at the Facility. The two byproducts in the event of a UF_6 release are hydrogen fluoride (HF) and uranyl fluoride (UO_2F_2), neither of which presents a process fire safety hazard. Chapter 3 for the ISA Summary identifies the Process Fire Safety Hazards. Thereby, the hazard events involving UF_6 and its byproducts are not directly applicable to the scope of the FHA or herein Chapter 7.

ISAS §2.5.2, Table 2-6 and Appendix 1 provide the controlled and uncontrolled accident sequences with the IROFS credited to achieve the acceptable risk levels of Table A-3 in NUREG-1520 (NRC, 2015). NFPA 401, *Recommended Practice for the Prevention of Fires and Uncontrolled Chemical Reactions Associated with the Handling of Hazardous Waste* (NFPA, 2024d) also provides guidance for fire and chemical event prevention that can apply to fuel cycle facilities.

Refer to License Application Chapters 3 and 6 for additional information on the OE Facility's process safety and the related hazards events that are evaluated by the ISA Summary.

7.5 FIRE PROTECTION AND EMERGENCY RESPONSE

This section documents the fire protection systems and emergency response organizations provided for the Facility.

7.5.1 Fire Protection System

The Facility fire protection systems consist of a dedicated fire water supply and distribution system, standpipe and hose systems, portable fire extinguishers, automatic suppression systems, fire detection system, and both manual (e.g., pull-box) and automatic alarm systems, following guidance of NFPA 72 (NFPA, 2019b).

7.5.1.1 Fire Water Supply and Distribution System

A Fire Water Supply System provides storage and distribution of water to fire protection features and systems that protect the Facility under the guidance of NFPA 22 (NFPA, 2023b).

A reliable fire protection water supply and distribution system of adequate flow, pressure, and duration following guidance of NFPA 24 (NFPA, 2025c) to meet the characteristics of the overall site, OE Facility and the scope of the FHA.

7.5.1.2 Standpipe and Hose Systems

As required by the FHA, standpipe systems and interior fire hose stations are provided and installed following guidance of NFPA 14 (NFPA, 2019c) in the following locations:

- Standpipe systems for FB and the offsite fire department use are provided in the stairwells of the Process Service Corridor of the SBMs and the stairwells in TSB and OSB.

The systems are designed to provide a minimum flow following guidance of NFPA 14 (NFPA, 2019c) for standpipe systems.

7.5.1.3 Portable Extinguishers

Portable fire extinguishers are installed throughout all buildings following guidance of NFPA 10 (NFPA, 2018). Multi-purpose extinguishers are provided in general areas for Class A, B, and C fires.

The portable fire extinguishers are spaced within the travel distance limitation and provide the area coverage following guidance of NFPA 10 (NFPA, 2018). Specialized extinguishers (e.g., Class D) are located in areas requiring protection from high temperature metal fires or require extinguishing fires without CO_2 or water. Supplemental fire extinguishers will be provided in water exclusion areas (e.g., OE Facility areas where sprinkler coverage would create additional criticality hazards). In fissile material areas where water discharge is prohibited due to NCS moderation controls, the preferred fire extinguishing agent is carbon dioxide due to its suitability for use on electrical equipment and its lack of being a hydrogenous moderator for criticality.

7.5.1.4 Automatic Suppression Systems

Fire sprinkler systems are engineered to protect specific hazards in accordance with parameters established by the FHA. NFPA 801 (NFPA, 2020b) provide guidance that fire sprinkler systems be provided for the nuclear related process areas of the Facility except where determined unnecessary or inappropriate by the FHA. For the Facility, there are areas where sprinklers may be omitted or only provide partial coverage due to the need to mitigate the risk of nuclear criticality.

In these cases, other controls to mitigate the impact of fire will be provided as required. The Facility FHA contains a methodology for comparative evaluation of fire and criticality risk. This methodology will be applied during detailed design to determine fissile material areas where sprinkler coverage should be limited, or omitted, and what other controls (e.g., alternate suppression, limitations on combustibles) should be applied.

Water flow detection is provided to annunciate all sprinkler alarm system actuations. Sprinkler system control valves are monitored using management measures under a periodic inspection program requirements and their proper positioning is supervised to ensure the systems remain operable.

7.5.1.5 Fire Detection Systems

Facility structures are provided with automatic fire detection installed following guidance of NFPA 72 (NFPA, 2019b) as required by the FHA and in accordance with the IBC (ICC, 2021). If a discrepancy within these consensus standards or programmatic documents is found, the more conservative approach will be selected within the best safety practices for the OE Facility. Automatic smoke, heat, or fire detectors are installed as appropriate to the hazard in all process structures as required by the FHA or in accordance with IBC (ICC, 2021) for early detection of fire conditions and/or the actuation of sprinklers to charge sprinkler supply piping within in the coverage areas of the fire system.

7.5.1.6 Manual Alarm Systems

All facility structures are provided with manual fire alarm pull stations installed following guidance of NFPA 72, (NFPA, 2019b), NFPA 101 (NFPA, 2021b); and the requirements of the FHA.

7.5.1.7 Fire Alarm System

Each building of the Facility is monitored by a local fire alarm control panel (LFACP) following guidance of NFPA 72 (NFPA, 2019b). Each panel has a dual power supply, consisting of normal and backup power. Activation of a fire detector, manual pull station, or water flow device results in audible and visual alarms at both the primary building and main fire alarm control panels.

The main fire alarm control panel (MFACP) are located in the Control Room, and connected to each individual LFACP. The MFACP is also fed by dual power sources that consist of both normal and backup power. The MFACP monitors all functions associated with the individual building alarm panels and the fire pump controllers. All fire alarms, suppression system actuation alarms, supervisory alarms, and trouble alarms are audibly and visually annunciated by the MFACP and automatically recorded. Failure of the MFACP will not result in failure of any building's LFACP and its associated local control functions (e.g., releasing or local alarming).

All fire pump alarm and trouble conditions are monitored by the MFACP through the fire pump controllers and annunciated following guidance of NFPA 20 (NFPA, 2019a).

7.5.2 Fire Emergency Response

7.5.2.1 Fire Brigade

The Facility maintains a FB made up of employees trained in fire prevention, firefighting techniques, first aid procedures, emergency response, and nuclear criticality safety. The FB is organized, operated, trained and equipped following guidance of NFPA 600 (NFPA, 2025e). The criticality safety training addresses water moderation, water reflection, product cylinder safety by moderation control, and mitigation/prevention of water flooding. The FB is considered an incipient FB, e.g., not required to wear thermal protective clothing nor self-contained breathing apparatus during firefighting. The intent of the OE Facility's FB is to respond and control minor fires, provide first response, and supplement the offsite fire department for any major fire at the Facility. The FB members are trained and equipped to respond to all fire emergencies and take first response actions until the local, offsite fire department arrives. First response firefighting by the FB includes using portable hand or wheeled fire extinguishers, advancing hose lines to fight interior/exterior incipient fires, and to fight larger exterior fires in a defensive mode (e.g., vehicle fires). FB members will not perform firefighting where conditions warrant firefighting gear.

When the local offsite fire department arrives onsite, the local fire department assumes control and is responsible for all firefighting activities. The Facility FB, working with the Facility Emergency Operations Center, will coordinate offsite fire department activities to ensure moderator control and criticality safety are maintained during firefighting activities.

Periodic training is provided to offsite emergency personnel in the Facility emergency planning procedures. Please refer to the Facility Emergency Management (EM) Plan for more information of the OE Facility Fire Brigade, their interface with the local offsite emergency personnel, and their training requirements.

7.5.2.2 Offsite Organizations

OE will use the services of local, offsite fire departments to supplement the capability of the Facility FB. Please refer to the Facility EM Plan for more information on the roles of local, offsite emergency personnel.

7.5.3 Physical Security Concerns

Detailed pre-fire plans provide on-site and off-site fire fighters a reference of materials and important features available for each area of the Facility. The pre-fire plans will include the location of fire protection equipment; approach paths for fire response and egress/access points.

The Physical Security Plan (PSP) is a blueprint for safeguarding an organization's critical assets against various threats, including theft, vandalism, terrorism, and espionage. It outlines the necessary layered protective measures and protocols to ensure a secure environment for personnel and property. Key aspects are (1) Threat Assessment, (2) Access Control, (3) Surveillance, (4) Incident Response, (5) Periodic Assessments for Effectiveness, (6) Programmatic Updates, and (6) Program Training. This plan provides access and egress points for onsite and offsite emergency response personnel.

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Chapter 8 – Emergency Management

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8.0 EMERGENCY MANAGEMENT

The plans for coping with emergencies are presented in the Facility Emergency Plan, developed in accordance with 10 CFR 70.22(i) and 10 CFR 40.31(j). The Emergency Plan conforms to the guidance presented in Regulatory Guide 3.67, Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities, (NRC 2011) and addresses the specific acceptance criteria in NUREG-1520 (NRC 2015), Chapter 8, Emergency Management.

The Emergency Plan identifies the offsite organizations that reviewed the plan pursuant to the requirement in 10 CFR 70.22(i)(4) and 10 CFR 40.31(j)(4). Memoranda of Understanding with the off-site organizations are also provided in the Emergency Plan.

8.1 REFERENCES

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Chapter 9 – Environmental Protection

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9.0 ENVIRONMENTAL PROTECTION

This section documents the potential environmental impacts associated with construction and operation of the Orano Enrichment USA LLC (OE) uranium enrichment facility, hereafter referred to as the "Facility", and indicates that adverse impacts are small. The Facility will meet the underlying need for additional reliable and economical uranium enrichment capacity in the United States, thereby serving important energy and national security policy objectives. Accordingly, because the impacts of the Facility are minimal and acceptable, and the benefits are desirable, the no-action alternative may be rejected in favor of the proposed action.

9.1 ENVIRONMENTAL REPORT

OE's prepared Environmental Report (ER) presents the proposed action, purpose of the proposed action, and applicable regulatory requirements in Chapter 1, discusses alternatives in Chapter 2, describes the facility and the affected environment in Chapter 3, and describes potential impacts of the proposed action in Chapter 4. Mitigation measures are described in Chapter 5, environmental measurements and monitoring programs in Chapter 6, a cost-benefit analysis in Chapter 7, and a summary of environmental consequences in Chapter 8. References and Contributors are listed in Chapters 9 and 10, respectively.

9.1.1 Environmental Considerations

OE's ER adequately addresses the requirements of 10 CFR 51.45(b) as follows.

9.1.1.1 Description of Proposed Action

The proposed action, described in ER Section 1.2 and 2.1.2, is the issuance of an NRC specific license under 10 CFR 30, 10 CFR 40 and 10 CFR 70 to possess and use byproduct material, source material and special nuclear material (SNM) and to construct and operate a uranium enrichment facility in Oak Ridge, Tennessee.

ER Chapter 1 includes the proposed project schedule. Significant characteristics of the facility, including major site features such as Facility design and operating parameters are described in ER Chapter 2 and Chapter 3. A discussion of how the SNM, in this case uranium hexafluoride (UF_6), will be processed to produce enriched uranium-235 (^{235}U) is described in ER Section 2.1.

9.1.1.2 Purpose of Proposed Action

ER Chapter 1 demonstrates the need for the facility. The demonstration provides the:

- Quantities of SNM used for domestic benefit
- A projection of domestic and foreign requirements for services
- Alternative sources of supply for OE's proposed services.

ER Chapter 1 and Chapter 2 discuss the effects if the Facility is not constructed, such as the effects to the nation's energy program.

9.1.1.3 Description of the Affected Environment

Chapter 3 of the ER contains detailed descriptions of the affected environment. The chapter provides a baseline characterization of the site and its environs prior to any disturbances associated with construction or operation of the facility. The following topics and corresponding ER chapter section include:

- Site location (including longitude and latitude) and facility layout (1.0)
- Regional demography (3.10) and land use (3.1)
- Socioeconomic information (3.10)
- Regional historic (3.8), archeological (3.8.), architectural (3.9), scenic (3.9), cultural (3.8), and natural landmarks (3.9)
- Local meteorology and air quality (3.6)
- Local surface water and ground water hydrology (3.4)
- Regional geology and seismology (3.3)
- Local terrestrial and aquatic ecology (3.5)

The baseline descriptions presented are from the most current information available. It was gathered from federal, state, and county sources along with on-site data. Therefore, the information represents both seasonal and long-term environmental trends.

9.1.1.4 Discussion of Considerations

ER Chapter 4 details environmental and socioeconomic effects due to site preparation and facility construction and operation. Chapter 2 describes alternatives to the proposed action, including siting and designs. Chapter 7 provides a discussion of the costs and benefits for each alternative as well as the relationship between short-term use and long-term productivity of the environment, and resources committed. In addition, Chapter 8 provides a summary of environmental consequences from all actions. The associated regulatory criteria and corresponding ER section are as follows:

9.1.1.4.1 Impact of the Proposed Action on the Environment

- Effects of site preparation and construction on land and water use (4.1 and 4.4)
- Effects of facility operation on human population – including consideration of occupation and public radiation exposure – and important biota (4.10 and 4.12)
- Any irreversible commitments of resources because of site preparation and facility construction and operation (4.1, 7.0, and 8.2)
- Environmental effects of the transportation of radioactive materials to and from the site (4.2)
- Environmental effects of accidents (4.12)
- Impacts on air and water quality (4.6 and 4.4)
- Impacts on cultural and historic resources (4.8)

9.1.1.4.2 Adverse Environmental Effects

Refer to Section 9.1.4 below for details on the associated ER chapters and topics.

9.1.1.4.3 Alternatives to the Proposed Action

ER Chapter 2 provides a complete description of alternatives to the proposed action. Included are the no action alternative scenarios as well as the siting criteria and technical design requirements in sufficient detail to allow a fair and reasonable comparison between the alternatives.

9.1.1.4.4 Relationship between Short- and Long-term Productivity

ER Chapter 7 includes the consideration of the short-term uses and productivity of the site during the active life of the facility. No adverse impacts on the long-term productivity of the environment after decommissioning of the facility have been identified. The European experience at the Almelo enrichment plant demonstrates that a centrifuge technology site can be returned to a greenfield site for use without restriction.

9.1.1.4.5 Irreversible and Irretrievable Commitments of Resources

Irreversible environmental commitments and irretrievable material resources also are included in the cost-benefit analysis in ER Chapter 7. They are part of the capital costs associated with the land and facility and operating and maintenance costs. No significant commitments are involved with the proposed action. The site should be available for unrestricted use following decommissioning. Some components may be reused or sold as scrap during the Facility life or following decommissioning.

9.1.2 Analysis of Effects of Proposed Action and Alternatives

ER Chapter 2 discusses the analysis of effects of the proposed action and alternatives in accordance with 10 CFR 51.45(c). The analysis considers and balances the environmental effects of the proposed action and alternatives available to reduce or avoid both environmental and socioeconomic effects and other benefits of the proposed action.

9.1.3 Status of Compliance

ER Section 1.3 summarizes the applicability of environmental regulatory requirements, permits, licenses, or approvals as well as the current status of each on the effective date of the ER.

Many federal laws and regulations apply to the facility during site assessment, construction, and operation. Some of these laws require permits from consultations with, or approvals by, other governing or regulatory agencies. Some apply only during certain phases of facility development, rather than the entire life of the facility. Federal statutes and regulations (non-nuclear) have been reviewed to determine their applicability to the facility site assessment, construction, and operation.

9.1.4 Adverse Information

In accordance with 10 CFR 51.45(e), various sections throughout the ER discuss adverse environmental effects. In particular, Chapter 4 details environmental and socioeconomic effects due to site preparation and facility construction and operation. Chapter 2 compares potential impacts from alternatives. Lastly, Chapter 8 provides a summary of environmental consequences from all actions.

9.2 ENVIRONMENTAL PROTECTION MEASURES

OE is committed to protecting the public, Facility workers, and the environment from the harmful effects of ionizing radiation due to Facility operation. Accordingly, OE is firmly committed to the As Low As Reasonably Achievable (ALARA) philosophy for all operations involving source, byproduct, and special nuclear material. This commitment is reflected in written procedures and instructions for operations involving potential exposures of personnel to radiation (both internal and external hazards) and the facility design. Written procedures for effluent monitoring address the need for periodic (monthly) dose assessment projections to members of the public to ensure that potential radiation exposures are kept ALARA (i.e., not in excess of 0.1 mSv/yr (10 mrem/yr)) in accordance with 10 CFR 20.1101(d).

Parts of OE's environmental protective measures are described in the ER. In particular, Chapter 4 discusses the anticipated results of the radiation protection program with regard to ALARA goals and waste minimization. Chapter 6 discusses the environmental controls and monitoring program.

A description of OE's radiation protection program is included in Chapter 4. Similarly, OE's provisions for a qualified and trained staff, which also is part of the environmental protection measures required, are established by the personnel qualifications of the management and supervisory staff as well as formal training for facility employees, as described in Chapter 2.

9.2.1 Radiation Safety

The four acceptance criteria that describe the facility radiation safety program are as follows:

- Effluent controls to maintain public doses ALARA

  The equipment and design features incorporated in the Facility are selected to keep the release of gaseous and liquid effluent contaminants as low as practicable, and within regulatory limits. They are also selected to minimize the use of depletable resources.

  Additional information is in Section 4.13 of the ER.

- Waste minimization.

  A high priority will be assigned to minimizing the generation of waste through reduction, reuse, or recycling. The Facility incorporates several waste minimization systems in its operational procedures that aim at conserving materials and recycling important compounds.

  OE has a program for pollution prevention and waste minimization that includes the following:

  - Waste minimization, reduction, reuse, and recycling for the various phases of the Facility construction and operation and decommissioning
  - Employee training and education on general environmental activities and hazards regarding the facility, operations and the pollution prevention program
  - Responsibilities for pollution prevention and waste minimization

  ALARA controls and BMPs will be maintained during facility operation to minimize the generation of radioactive waste as directed in 10 CFR 20.

ER Section 4.12 describes public and occupational health effects from both non-radiological and radiological sources. This section specifically addresses calculated total effective dose equivalent (TEDE) to an average member of critical groups or calculated average annual concentration of

radioactive material in gaseous and liquid effluent to maintain compliance with 10 CFR 20. ER Section 4.13 contains information on facility waste minimization.

License Application Chapter 4 describes:

- Radiological (ALARA) goals for effluent control
- ALARA reviews and reports to management.

9.2.2 Effluent and Environmental Controls and Monitoring

OE will design an environmental monitoring program to provide comprehensive data to monitor the facility's impact on the environment. The preoperational program will focus on collecting data to establish baseline information useful in evaluating potential changes in environmental conditions caused by facility operation. The preoperational program will be initiated at least two years prior to facility operation.

The operational program will monitor to ensure facility emissions are maintained in accordance with ALARA principles and meet the air permit requirements from Tennessee Department of Environment and Conservation (TDEC). Sampling focuses on locations within the site perimeter but may also include distant locations as control sites. Sampling locations have been determined based on NRC guidance found in NUREG-1302 (NRC 1991), meteorological information, and current land use. In conjunction with the existing DOE-Legacy sample monitoring wells already at the site, additional sampling locations may be subject to change as determined from the results of periodic review of land use.

ER Chapter 6 describes the environmental measurement and monitoring programs as they apply to pre-operation (baseline), operation, and decommissioning conditions for both the proposed action and each alternative.

9.2.2.1 Effluent Monitoring

ER Section 6.1 presents information relating to the facility radiological monitoring program. This section describes the location and characteristics of radiation sources and radioactive effluent (Figure 9-1).

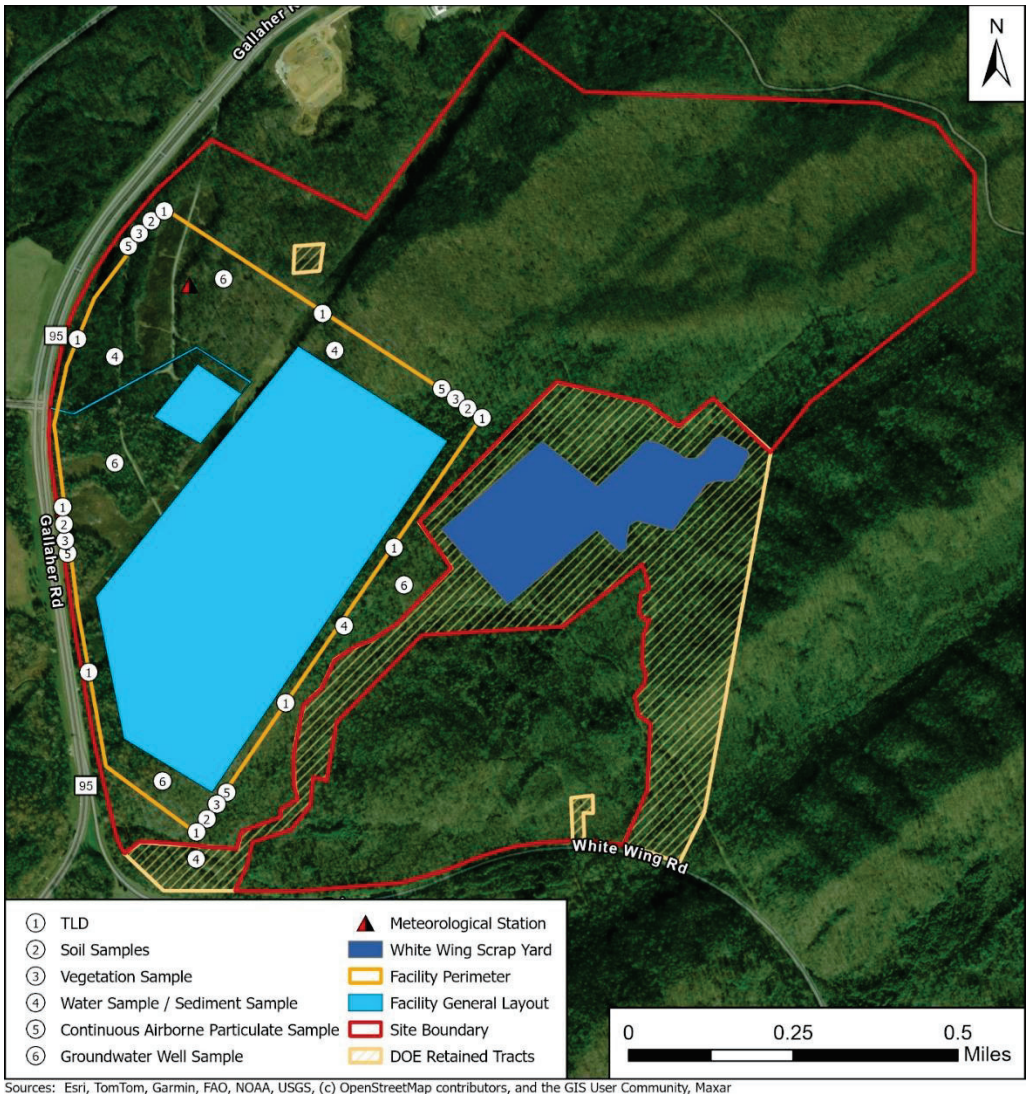

Figure 9-1 Proposed Facility Monitoring Locations

It also describes the various elements of the monitoring program, including:

- Number and location of sample collection points
- Measuring devices used
- Pathway sampled or measured
- Collection frequency and duration
- Method and frequency of analysis, including lower limits of detection.

As a matter of compliance with regulatory requirements, all potentially radioactive effluent from the facility is discharged only through monitored pathways. See ER Section 4.12.3 for a discussion of pathway assessment. The effluent sampling program for the Facility is designed to determine the quantities and concentrations of radionuclides discharged to the environment. The uranium isotopes ^{238}U, ^{236}U, ^{235}U, and ^{234}U are expected to be the prominent radionuclides in the gaseous effluent. The annual uranium source term for routine gaseous effluent releases from the 7.4 million SWU Facility has been conservatively assumed to be less than 592 µCi per year, which

is proportional to the 120 µCi per year source term applied to the 1.5 million SWU Facility described in NUREG-1484 (NRC 1994).

This is a very conservative annual release estimate used for bounding analyses. Additional details regarding source term are provided in ER Section 4.12. Representative samples are collected from each release point of the facility. Because uranium in gaseous effluent may exist in a variety of compounds, effluent data will be maintained, reviewed, and assessed by the Facility's Industrial Safety & Radiation Protection Manager to assure that gaseous effluent discharges comply with regulatory release criteria for uranium. Table 9-1 presents an overview of the effluent sampling program.

The Facility has two primary gaseous effluent streams: the effluent discharged from the GEVS and the chemical and thermal gaseous effluent released through the HVAC systems.

The GEVS is designed to remove particulates containing uranium and hydrogen fluoride (HF) from process gas streams that may be contaminated. The system uses prefilters and high-efficiency particulate air (HEPA) filters, along with potassium hydroxide (KOH)-impregnated activated carbon filters, to capture radioactive particles and absorb HF before the gas is released to the exhaust.

The HVAC systems release the facility’s remaining gaseous effluents, including both chemical constituents and thermal waste, through their discharge streams.

A list of gaseous effluents and associated annual quantities (estimated) is presented in Section 4.13 of the Environmental Report.

There are no liquid effluent discharges from Facility operations. Radiologically contaminated, potentially contaminated, and non-radiologically contaminated aqueous liquid effluents are generated in various processes throughout the Facility. Most potentially radioactive aqueous effluents originate in the Technical Support Building (TSB), where all aqueous liquid streams are collected in tanks located in the Liquid Effluent Collection and Treatment Room. The collected liquids are sampled and analyzed to determine whether treatment is required prior to atmospheric release by evaporation.

When treatment is necessary, the effluent undergoes filtration and precipitation processes to remove uranium and fluorine. These cycles are repeated as needed until concentrations reach acceptable levels, after which the liquid is transferred to an evaporator for vaporization and final discharge to the atmosphere.

Non-contaminated aqueous effluents are also collected and monitored. If they meet all regulatory and administrative criteria, they are evaporated and discharged directly to the atmosphere.

Estimated annual quantities of liquid effluents are provided in Section 4.13 of the Environmental Report.

Table 9-1 Effluent Monitoring Program

Sample Location	Sample Type	Analysis / Frequency
Buildings GEVS exhaust vents	Continuous air particulate filter	Gross alpha/beta-Weekly Isotopic analysis[4]-Quarterly composite
Process Areas[2]	Local area continuous air particulate filter[3]	Gross alpha/beta-Weekly Isotopic analysis[4]-Quarterly composite
Non-Process Areas[2]	Local area continuous air particulate filter[3]	Gross alpha/beta-Quarterly composite

Notes:

1. The continuous sampling system is operated only when the Centrifuge Test Facility or Postmortem Facility is in operation.
2. A "Process Area" is any area of the facility where UF_6 process flow between feed, product, or tails cylinders occurs, including areas where cylinders containing UF_6 are opened for testing, inspection, or sampling. A "Non-Process Area" is any other area where UF_6 is present only in fully contained form.
3. These will generally be collected with mobile continuous air monitors, as required to complement the effluent monitoring program.
4. Isotopic analysis for Uranium if gross alpha and gross beta activities indicate that an individual radionuclide could be present in a concentration greater than 10 percent of the concentrations specified in Table 2 of Appendix B to 10 CFR Part 20.

9.2.2.1.1 Expected Concentrations

Pursuant to 10 CFR 20, surveys necessary to demonstrate compliance with these regulations and to demonstrate that the amount of radioactive material present in effluent from the facility has been kept ALARA are required. In addition, the NRC has issued Regulatory Guide 4.15 (NRC 2007) and Regulatory Guide 4.16 (NRC 2010) that reiterate that concentrations of hazardous materials in effluent must be controlled and that licensees must adhere to the ALARA principal such that there is no undue risk to the public health and safety at or beyond the site boundary.

As noted in ER Section 6.1.1, discharge of gaseous effluent has the highest possibility of the potential pathways, of introducing facility-related uranium into the environment. However, the radioactive materials in gaseous effluents from the Facility are expected to be very low concentrations of uranium because of process and effluent controls. Under routine operating conditions, radioactive material in effluents discharged from the facility will comply with regulatory release criteria.

9.2.2.1.2 Calculation of Total Effective Dose Equivalent

Based on recorded Facility effluent data, dose projections to members of the public will be performed as required to ensure that the annual dose to members of the public does not exceed the ALARA constraint of 0.1 mSv/yr (10 mrem/yr) from air emissions and radioactive materials. Compliance is demonstrated through effluent and environmental sampling data. Compliance with 10 CFR 20.1301 will be demonstrated using a calculation of the TEDE to the individual who is likely to receive the highest dose in accordance with 10 CFR 20.1302(b)(1). Pursuant to 10 CFR 70, semiannual reports will be submitted, specifying the quantities of the principal radionuclides released to unrestricted areas and other information needed to estimate the annual radiation dose to the public from effluent discharges. If the monthly dose impact assessment indicates a trend in effluent releases that, if not corrected, could cause the ALARA constraint to be exceeded, appropriate corrective action will be initiated to reduce the discharges to assure that subsequent

releases will be in compliance with the annual dose constraint. In addition, an evaluation of the need for increased sampling will be performed.

9.2.2.1.3 Effluent Discharge Locations and Sampling

Effluents will be sampled as indicated in Table 9-1, which presents an overview of the effluent sampling program. For gaseous effluents, liquid condensate samples from the evaporator exhaust vent and continuous air sampler filters are analyzed for gross alpha and gross betas required. The filters, or liquid condensate samples, are composited as required and an isotopic analysis is performed if a specified gross alpha or gross beta action level is exceeded (as specified in Table 9-1). Table 9-2 summarizes detection requirements for gaseous effluent sample analyses. Sampling of liquid effluent discharges to the detention and retention basins are described below in Section 9.2.2.2.

Table 9-2 Required Lower Limit of Detection for Effluent Sample Analysis

Effluent Type	Nuclide	MDC in Bq/mL (µCi/mL)
Gaseous[1]	Isotopic U	5.6×10^{-9} (1.5×10^{-13})
Gaseous[1]	Gross Alpha	5.6×10^{-9} (1.5×10^{-13})
Liquid[2]	Isotopic U	5.6×10^{-4} (1.5×10^{-8})

Notes:

1. These MDCs are 5% of the limits in 10 CFR 20 Appendix B, Table 2 Effluent Concentrations (retention Class D for Air)
2. This MDC is 5% of limit in 10 CFR 20 Appendix B, Table 2, Col. 2 (Water) value for retention Class D.

The guidance in NUREG-1302 (NRC 1991) and Regulatory Guide 4.16 (NRC 2010) were followed for determining sample locations, analyses, frequencies, durations, and lower limits of detection.

Lastly, Chapter 6 of the ER justifies the choice of sample locations, analyses, frequencies, durations, and lower limits of detection.

9.2.2.2 Environmental Monitoring

As described in ER Chapter 6, an environmental monitoring program is to be implemented to measure non-radiological chemical impacts upon the natural environment.

The environmental monitoring program relies on the data acquisition from the radiological and physiochemical monitoring programs to detect and contain any potentially adverse releases from the Facility to the environment.

Sampling locations are determined based on meteorological information and current land use. The sampling locations may be subject to change as determined from the results of any observed changes in land use.

The range of chemical surveillance incorporated into all the planned monitoring programs are designed to be sufficient to predict any relevant chemical interactions in the environment related to Facility operations.

Environmental monitoring program activities would include periodic flora and fauna surveys of the land to identify potential problems and areas needing improvement

The Radiological Environmental Monitoring Program (REMP) at the Facility is a major part of the effluent compliance program. It provides a supplementary check of containment and effluent controls, establishes a process for collecting data for assessing radiological impacts on the environs and estimating the potential impacts on the public, and supports the demonstration of compliance with applicable radiation protection standards and guidelines. The REMP includes the collection of data during pre-operational years to establish baseline radiological information that will be used in determining and evaluating impacts from operations at the Facility on the local environment (Table 9-3). The REMP will be prior to Facility operations in order to develop a sufficient baseline. The early initiation of the REMP provides assurance that a sufficient environmental baseline has been established for the Facility before the arrival of the first uranium hexafluoride shipment.

A map of ecological sample locations is shown in Figure 9-2.

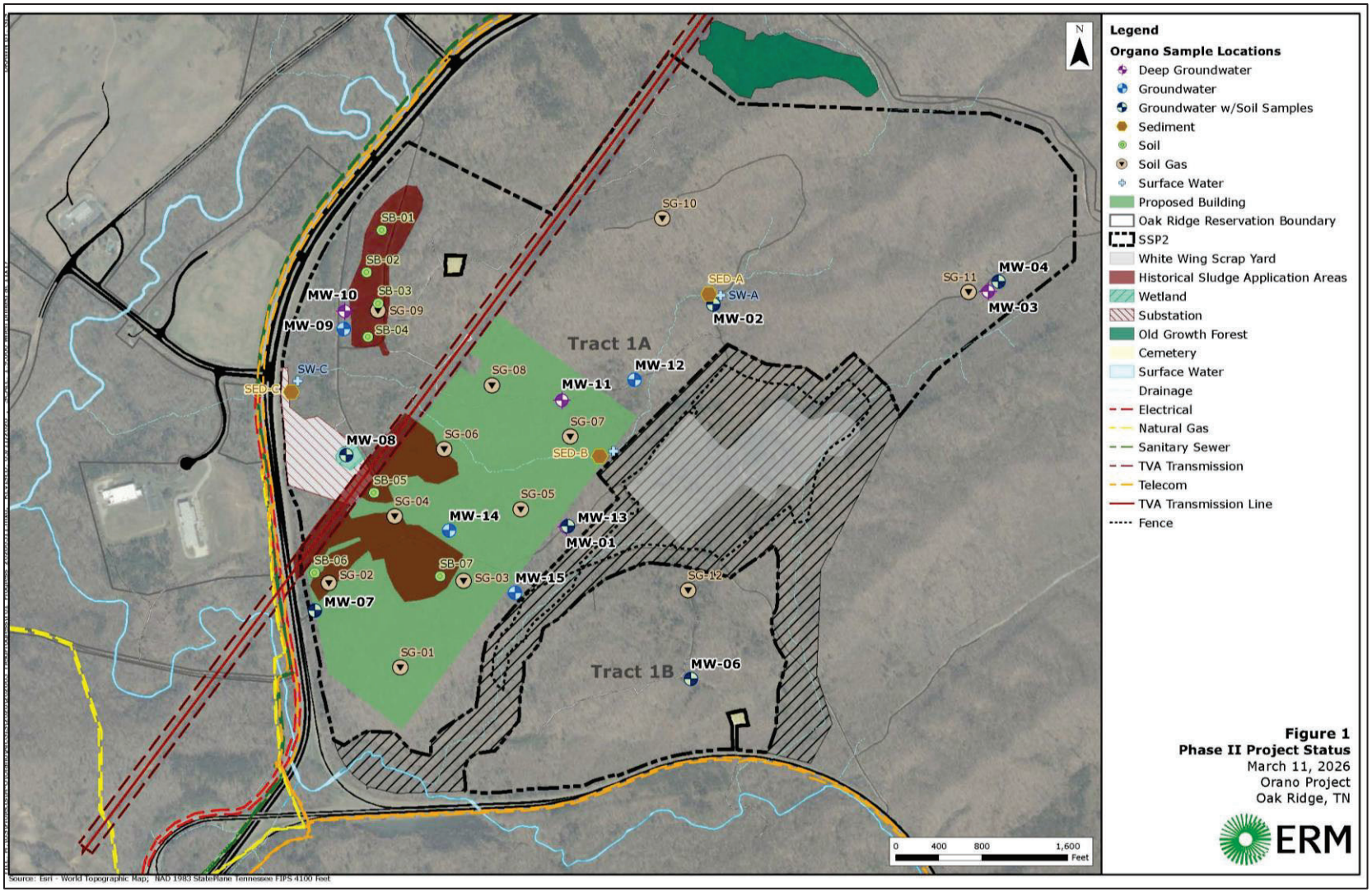

Figure 9-2 Ecological Sampling Locations

Table 9-3 Preoperational Radiological Environmental Monitoring Program

Sample Type	Location	Minimum Number of Sample Locations	Sampling and Collection Frequency	Type of Analysis
Continuous Airborne Particulate	Note 4	4	Continuous operation of air sampler with sample collection as required by dust loading but at least biweekly. Quarterly composite samples by location.	Gross beta/gross alpha analysis each filter change. Quarterly isotopic analysis on composite sample.
Vegetation	Note 4	5 5	1 to 2-kg (2.2 to 4.4-lb) samples collected semiannually Quarterly if present (i.e., during growing seasons); one sample at each location	Isotopic analysis [1] Fluoride uptake
Groundwater	Note 4	4	4-L (1.06-gal) samples collected semiannually	Isotopic analysis [1]
Basins	Note 2	1 from each of 2 basins [2] Discharge points to the basins [2]	4-L (1.06-gal) water sample/1 to 2-kg (2.2 to 4.4-lb) sediment sample collected quarterly Quarterly for one sediment sample at each location	Isotopic analysis [1] Fluoride uptake
Soil	Note 4	4	1 to 2-kg (2.2 to 4.4-lb) samples collected semiannually Quarterly, near vegetation sample locations; one sample at each location	Isotopic analysis [1] Fluoride uptake
Domestic Sanitary Sewage Treatment Plant [3]	Note 3	Note 3	Note 3	Note 3
TLD	Note 4	9	Quarterly	Gamma and neutron dose equivalent

Notes:

1. Isotopic analysis for Uranium.
2. Site Stormwater Detention Basins and Cylinder Storage Pads Stormwater Retention Basins.
3. Sanitary wastewater from the DSSTP is not expected to be contaminated with licensed material. Therefore, OE does not sample or analyze the untreated sewage
4. See legend of Figure 9-1 and Figure 9-2 where each sample type will be collected.

A minimum detectable concentration (MDC) of at least $1.5x10^{-13}$ µCi/mL is a program requirement (NRC 2015) for all analyses performed on gaseous effluent samples. That MDC value represents 5% of the limit for any applicable uranium isotope (Class D). Liquid condensate samples from the evaporator discharge are analyzed to an MDC equivalent to 5% or less of the appropriate 10 CFR 20 Appendix B, Table 2, Col. 1 (ambient air) value. The MDCs for gross alpha (assumed to be uranium) in various environmental media are shown in Table 9-4.

Table 9-4 Required MDC for Environmental Sample Analysis

Medium	Analysis	MDC Bq/mL or g (µCi/mL or g)
Ambient Air[1]	Gross Alpha	2.2×10^{-9} (6.0×10^{-14})
Vegetation	Isotopic U	1.9×10^{-4} (5.0×10^{-9})
Soil/Sediment	Isotopic U	1.1×10^{-2} (3.0×10^{-7})
Groundwater[2]	Isotopic U	2.2×10^{-4} (6.0×10^{-9})

Notes:

1. MDCs are 2% or less of the limits in 10 CFR 20 Appendix B, Table 2 Effluent Concentrations (retention Class D for ambient air).
2. MDCs are 2% or less of the limits in 10 CFR 20 Appendix B, Table 2 Effluent Concentrations (retention Class D for water).

9.2.2.3 Waste Minimization

A high priority will be assigned to minimizing the generation of waste through reduction, reuse, or recycling. The equipment and design features incorporated in the Facility are selected to keep the release of gaseous and liquid effluent contaminants as low as practicable, and within regulatory limits. They are also selected to minimize the use of depletable resources.

The Facility design serves to minimize the use of depletable resources. Water is the primary depletable resource used at the Facility. Electric power usage also depletes fuel sources used in the production of the power.

During operation, a non-hazardous materials waste recycling plan will be implemented. The recycling plan will start with the performance of a waste assessment to identify waste reduction opportunities and to determine which materials will be recycled.

The Facility will implement a spill control program for accidental oil spills. Its purpose will be to reduce the potential for the occurrence of spills, reduce the risk of injury if a spill occurs, minimize the impact of a spill, and provide a procedure for the cleanup and reporting of spills. The oil spill control program will be established to comply with the requirements of 40 CFR 112.

ALARA controls will be maintained during facility operation to minimize the generation of radioactive waste as directed in 10 CFR 20.

A full description of the waste impacts is discussed in Section 4.13 of the Environmental Report (OE 2026).

9.2.2.4 Data Analysis

Written procedures will be in place to ensure the collection of representative samples, use of appropriate sampling methods and equipment, proper locations for sampling points, and proper handling, storage, transport, and analyses of effluent samples. In addition, the Facility's written procedures also ensure that sampling and measuring equipment, including ancillary equipment such as airflow meters, are properly maintained and calibrated at regular intervals. Sampling equipment (pumps, pressure gages, and air flow calibrators) will be calibrated by qualified individuals. Sampling equipment and lines will be inspected for defects, obstructions, and cleanliness. Calibration intervals will be developed based on applicable industry standards.

9.2.2.5 Laboratory Quality Control

All environmental samples will be analyzed by an accredited offsite laboratory. Monitoring and sampling activities, laboratory analyses, and reporting of Facility-related radioactivity in the environment will be conducted in accordance with industry-accepted and regulatory-approved methodologies.

9.2.2.6 Action Levels

Administrative action levels are established for effluent samples and monitoring instrumentation as an additional step in the effluent control process. All action levels are sufficiently low so as to permit implementation of corrective actions before regulatory limits are exceeded.

As noted in ER Chapter 6, administrative action levels are established for effluent samples and monitoring instrumentation as an additional step in the effluent control process. All action levels are sufficiently low so as to permit implementation of corrective actions before regulatory limits are exceeded. Effluent samples that exceed the action level are cause for an investigation into the source of elevated radioactivity. Administrative action levels will be implemented prior to Facility operation to ensure that chemical discharges remains below the limits specified in the Facility discharge permits.

9.2.2.7 Federal and State Standards for Discharges

ER Section 1.3 describes all applicable federal and Tennessee state standards for discharges, as well as required permits issued by local, state, and Federal governments.

9.2.2.8 Reporting

Radiological reporting procedures will comply with the requirements of 10 CFR 70.59 and the guidance specified in Regulatory Guide 4.16 (NRC 2010). Reports of the concentrations of principal radionuclides released to unrestricted areas in effluents will be provided and will include the MDC for the analysis and the error for each data point. Each year, the Facility will submit a summary report of the environmental sampling program to the NRC, including all associated data as required by 10 CFR 70. The report will include the types, numbers, and frequencies of environmental measurements and the identities and activity concentrations of facility-related radionuclides found in environmental samples, in addition to the MDC for the analyses and the error associated with each data point.

9.2.3 Integrated Safety Analysis

OE has prepared an Integrated Safety Analysis (ISA) in accordance with 10 CFR 70.60. The ISA:

- Provides a complete list of the accident sequences that if uncontrolled could result in radiological and non-radiological releases to the environment with intermediate or high consequences.
- Provides reasonable estimates for the likelihood and consequences of each accident identified.
- Applies acceptable methods to estimate environmental effects that may result from accidental releases.
- Identifies adequate engineering and/or administrative controls for each accident sequence of environmental significance
- Assures adequate levels are afforded so those items relied on for safety (IROFS) will satisfactorily perform their safety functions.

The ISA demonstrates that the facility and its operations have adequate engineering and/or administrative controls in place to prevent or mitigate high and intermediate consequences from the accident sequences identified and analyzed. Corresponding management measures are described in Chapter 11.

9.3 PHYSIOCHEMICAL MONITORING

Monitoring procedures will employ well-known, acceptable analytical methods and instrumentation. The instrument maintenance and calibration program will comply with manufacturer recommendations. Environmental personnel at the proposed EREF will follow certified sampling and analysis protocols and implement appropriate steps to make sure that the onsite laboratory and any contractor laboratories participate in third-party laboratory inter-comparison programs appropriate to the media and parameters being measured.

The radiological environmental laboratory areas are located in the Technical Support Building (TSB). The non-radiological Environmental Laboratory areas are located in the Operation Support Building (OSB) and are used to perform analyses that include the following:

- Hazardous material presence in waste samples
- pH, oil and other contaminants in liquid waste streams

The environmental laboratory areas will be available to perform analyses on air, water, soil, and flora samples obtained from designated areas around the plant.

In addition to its environmental and radiological capabilities, the capability exists to perform bioassay analyses when necessary. Commercial, offsite laboratories may also be contracted to perform bioassay analyses.

All waste liquids, solids and gases from enrichment-related processes and decontamination operations will be analyzed and/or monitored for chemical and radiological contamination to determine safe disposal methods and/or further treatment requirements.

See Environmental Report Chapter 6 for a description of the physiochemical monitoring program. Below are the physiochemical sampling information (Table 9-5) and Stormwater Monitoring (Table 9-6) information tables.

Table 9-5 Physiochemical Sampling

Media	Number of Locations	Monitoring Frequency	Sample Type	Analysis[1]
Groundwater	5 deep wells and 10 shallow well used for baseline monitoring.	Semiannually for deep wells; semiannually for shallow wells when water is present	Grab	Metals, organics and pesticides; water level elevations
Soil[2]/sediment	3 minimum soil samples at locations to be determined by environmental staff plus one at each of the two stormwater retention basin outfalls.	Quarterly, near vegetation sample locations; one sample at each location Quarterly for one sample at each location	Surface grab Surface grab	Metals, organics, pesticides and fluoride uptake Metals, organics, pesticides and fluoride uptake
Surface water[2]	Potential location in intermittent stream drainage on southwestern corner of site.	Quarterly if water present	Grab	Metals, organics and pesticides
Stormwater[2]	Retention basins at locations to be determined by environmental staff.	Quarterly if water present	Grab	See Table 9-6
Vegetation[2]	4 minimum	Quarterly if present (i.e., during growing seasons); one sample at each location	Surface grab	Fluoride uptake

Notes:

[1] Analyses will meet EPA Lower Limits of Detection (LLD), as applicable, and will be based on the baseline surveys and the type of matrix (sample type).

[2] Location to be established by Environmental, Health, Safety and Licensing (EHS&L) organization staff.

Table 9-6 Stormwater Monitoring Program for Retention Basins [1]

Monitored Parameter	Monitoring Frequency	Sample Type	LLD[2] (ppm)
Oil and Grease	Quarterly, if standing water exists	Grab	0.5
Total Suspended Solids	Quarterly, if standing water exists	Grab	0.5
Five-Day Biological Oxygen Demand	Quarterly, if standing water exists	Grab	2
Chemical Oxygen Demand	Quarterly, if standing water exists	Grab	1
Total Phosphorus	Quarterly, if standing water exists	Grab	0.1
Total Kjeldahl Nitrogen	Quarterly, if standing water exists	Grab	0.1
pH	Quarterly, if standing water exists	Grab	0.01 units
Nitrate plus Nitrite Nitrogen	Quarterly, if standing water exists	Grab	0.2
Metals	Quarterly, if standing water exists	Grab	Varies by metal

Notes:

[1] Site Stormwater Detention Basins, Cylinder Storage Pads Stormwater Retention Basins and any temporary basin(s) used during construction.

[2] Lower limit of detection; Analyses will meet EPA LLD, as applicable, and will be based on the baseline surveys and the type of matrix (sample type).

9.4 REFERENCES

NRC. 1994. "Final Environmental Impact Statement for the Construction and Operation of Claiborne Enrichment Center, Homer, Louisiana." *U.S. Nuclear Regulatory Commission* NUREG-1484 (Volume 1).

NRC. 2010. "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluent from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plan." *U.S. Nuclear Regulatory Commission* Regulatory Guide 4.16 (Revision 2).

NRC. 1991. "Off-site Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors." *U.S. Nuclear Regulatory Commission* NUREG-1302.

NRC. 2007. "Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment." *U.S. Nuclear Regulatory Commission* Regulatory Guide 4.15 (Revision 2).

NRC. 2015. "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." *U.S. Nuclear Regulatory Commission* NUREG-1520 (Revision 2).

OE. 2026. *Orano Enrichment USA LLC Enrichment Facility Environmental Report.* Environmental Report, Orano Enrichment USA LLC, U.S. Nuclear Regulatory Commission. https://www.nrc.gov/docs/ML2603/ML26030A236.html.

Chapter 10 – Decommissioning

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10.0 DECOMMISSIONING

In accordance with 10 CFR 70.22(a)(9), this chapter discusses conceptual approach for meeting the decommissioning requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, Subpart E. A general description of decommissioning costs and explanation of the arrangements made to assure funding is available to cover these costs is also provided. A Decommissioning Funding Plan (DFP) in accordance with 10 CFR 70.25(b), prepared in accordance with NUREG-1757 (NRC 2012) accompanies this chapter.

10.1 CONCEPTUAL DECONTAMINATION AND DECOMMISSIONING PLAN

Decommissioning planning begins with ensuring design features are incorporated into the plant's initial design that will simplify eventual dismantling and decontamination. The plans are implemented through proper management and health and safety programs. Decommissioning policies address radioactive waste management, physical security, and material control and accounting.

The Facility is designed and operated in accordance with 10 CFR 20.1406 to minimize contamination, facilitate eventual decommissioning, and minimize to the extent practicable, the generation of radioactive waste. As a result, worker exposure to radiation and radioactive waste volumes during operations and decommissioning are maintained as low as reasonably achievable (ALARA).

In accordance with 10 CFR 30.35, 10 CFR 40.36, and 10 CFR 70.25, a DFP is submitted concurrent with the license application that contains a cost estimate for decommissioning and a description of the method of assuring funds for decommissioning. The DFP is prepared consistent with the guidance in NUREG-1757 (NRC 2012).

Orano Enrichment USA LLC (OE) commits to decontaminate and decommission the Facility and the site at the end of its operation so that the Facility and grounds can be released for unrestricted use. Prior to decommissioning, a Decommissioning Plan (DP) will be prepared in accordance with 10 CFR 30.36, 10 CFR 40.42 and 10 CFR 70.38 and submitted to the NRC for approval.

Facility Description

The Facility is fully described in other sections of this License Application. Information relating to the following topics can be found in the referenced chapters listed below:

- A general description of the Facility and plant processes is presented in Chapter 1.
- A description of the specific quantities and types of licensed materials used at the Facility is provided in Chapter 1.
- A general description of how licensed materials are used at the Facility is provided in Chapter 1.

10.1.1 Decommissioning Strategy

OE intends to decommission the Facility after shutdown, meaning the commencement of decontamination or removal of all materials from the site which prevent release of the Facility and site for unrestricted use. This approach, referred to in the industry as DECON (i.e., immediate dismantlement), avoids long-term storage and monitoring of wastes on site. The type and volume

of wastes produced at the Facility do not warrant delays in waste removal normally associated with the SAFSTOR (i.e., deferred dismantlement) option.

At the end of useful plant life, the Facility will be decommissioned such that the site and remaining facilities may be released for unrestricted use as defined in 10 CFR 20.1402. The intent of decommissioning the Facility is to remove all enrichment-related equipment from the buildings such that only the building shells and site infrastructure remain. Enrichment equipment will be removed, and all contaminated portions of the Facility will be decontaminated where needed to acceptable levels for unrestricted use. Requirements for Material Control and Accounting will be maintained during decommissioning in a manner similar to the programs in force during operation of the Facility.

Before decommissioning activities begin, an assessment of the radiological status of the Facility will be performed, and a DP will be prepared and submitted to the NRC pursuant to 10 CFR 70.38, Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas. The DP will provide information concerning the Facility, the types of equipment and areas to be decontaminated, the disposition of equipment used for hazardous materials, the assumptions upon which the cost of decommissioning is derived, and an estimated schedule for decommissioning and closing the Facility. Overall, decommissioning is estimated to require approximately 8 years from shutdown to completion of the final status survey of radiological conditions.

Depleted Uranium Hexafluoride (DUF_6), if not already sold or otherwise disposed of prior to decommissioning, will be disposed of in accordance with regulatory requirements. As described in Section 10.2.2, for cost estimate purposes, the DUF_6 will be transported to the Department of Energy (DOE) facilities at Portsmouth, Ohio or Paducah, Kentucky for conversion and disposal in accordance with regulatory requirements. Radioactive waste will be disposed of in licensed low-level radioactive waste disposal sites. Hazardous waste will be treated or disposed of in licensed hazardous waste facilities. No disposal of radioactive or hazardous material will occur at the plant site, but at licensed facilities located elsewhere.

Following decommissioning, no part of the facilities or site will remain restricted to any specific type of use.

Major features incorporated into the Facility design that facilitate decontamination and decommissioning are described below.

10.1.1.1 Radioactive Contamination Control

The following features primarily serve to minimize the spread of radioactive contamination during operation and therefore simplify eventual decommissioning. As a result, worker exposure to radiation and radioactive waste volumes are minimized as well.

- All areas of the plant are sectioned off into Unrestricted and Restricted Areas. Restricted Areas limit access for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Radiation Areas and Airborne Contamination Areas have additional controls to inform workers of the potential hazard in the area and to help prevent the spread of contamination. All procedures for these areas fall under the Radiation Protection Program and serve to minimize the spread of contamination and simplify the eventual decommissioning.
- Non-radioactive process equipment and systems are minimized in locations subject to potential contamination. This limits the size of the Restricted Areas and limits the activities occurring inside these areas.

10.1.1.2 Worker Exposure and Waste Volume Control

The following features primarily serve to minimize worker exposure to radiation and minimize radioactive waste volumes during decommissioning 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.
- Design drawings prepared for the Facility simplify the planning and implementing of decontamination procedures. This in turn will shorten the durations that workers are exposed to radiation.
- Worker access to contaminated areas is controlled to ensure that workers wear proper protective equipment and limit worker time in the areas.
- Remote equipment will be used when feasible.

10.1.2 Decommissioning Process

10.1.2.1 Overview

The four Separation Building Modules will be shut down in sequence. Since only low radiation levels exist at the Facility, decommissioning may begin immediately following the permanent shutdown of the first series of cascades in a Separation Building Module. The decommissioning of a single Separation Building Module is assumed to take 4.5 years; 3 years for decommissioning of the centrifuges and associated equipment and 1.5 years for decontamination of the structure. Dismantling and decontamination of the equipment in the four Separation Building Modules will be performed in a phased approach such that the decommissioning of all four Separation Building Modules is completed within an eight-year time frame.

Termination of the Facility operations will mark the end of uranium enrichment operations at the Facility. Also, decommissioning of the remaining plant systems and buildings will begin after the Facility operations have been permanently terminated.

Prior to beginning decommissioning operations, an extensive radiological survey of the Facility will be performed in conjunction with a historical site assessment. The findings of the radiological survey and historical site assessment will be presented in a Decommissioning Plan to be submitted to the NRC. The Decommissioning Plan will be prepared in accordance with 10 CFR 70.38 and the applicable guidance provided in NUREG-1757 (NRC 2012).

Decommissioning activities will generally include: (1) installation of decontamination equipment, (2) purging of process systems, (3) dismantling and removal of equipment, (4) decontamination and destruction of Classified material, (5) sales of salvaged materials, (6) disposal of wastes, and (7) completion of a final status survey.

Note:

- Credit is not taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning.
- Depending on technological developments occurring prior to Facility shutdown, the tails may have become marketable for further enrichment or other processes. The disposition of DUF_6 and relevant funding provisions are discussed in Section 10.2.2, DUF_6

Disposition. The cost estimate takes no credit for any value that may be realized in the future due to the potential marketability of the stored tails.

10.1.2.2 Installation of decontamination equipment

Decontamination of plant components and structures will require installation of new equipment dedicated for that purpose. Existing buildings, such as the Centrifuge Assembly Building, are assumed to house the equipment. These equipment will be specially designed to accommodate repetitive cleaning of thousands of centrifuges. The new equipment will be used for decontamination activities during the decommissioning process. The decontamination area in the Technical Support Building, used during normal operation, may also handle small items at decommissioning.

10.1.2.3 Shutdown and Purging

At the end of the useful life of each Separation Building Module, the enrichment process is shut down and UF_6 is removed to the fullest extent possible by normal process operation. This is followed by evacuation and purging with nitrogen. Connections in the process systems provided for required operation and maintenance allow for thorough purging at shutdown. This will remove a significant portion of radioactive contamination prior to disassembly. This shutdown and purging portion of the decommissioning process is estimated to take approximately three months.

10.1.2.4 Dismantling and Removal

Dismantling is the process of disassembling, disconnecting, or cutting of components requiring removal. The dismantling and removal activities are simple but labor intensive and generally require the use of protective clothing or equipment. The work process will be optimized considering the following:

- Minimizing the spread of contamination and the need for protective clothing or equipment
- Balancing the number of cutting and removal operations with the resultant decontamination and disposal requirements
- Optimizing the rate of dismantling with the rate of decontamination throughput
- Providing storage and laydown space required, as impacted by retrievability, radiation protection, criticality safety, and security and
- Balancing the cost of decontamination with the cost of disposal

Details of the optimization process will be decided near the end of the Facility useful life, considering specific contamination levels, market conditions, and available waste disposal sites. To avoid lay down space and contamination problems, dismantling should be coordinated with, and proceed generally no faster than, the downstream decontamination process.

10.1.2.5 Decontamination

The decontamination process is addressed separately in detail in Section 10.1.8.

10.1.2.6 Sale of Salvaged Equipment and Materials

Items to be removed from the facilities can be categorized as potentially re-usable equipment, recoverable scrap, and wastes. However, based on a 60-year facility operating useful life, operating equipment is not assumed to have reuse value. Wastes will also have no salvage value.

With respect to scrap, a significant amount of aluminum, along with smaller amounts of steel, copper, and other metals, may be recovered. For security and convenience, the uncontaminated materials will likely be smelted to standard ingots, and, if possible, sold at market price. The contaminated materials will be disposed of as low-level radioactive waste. No credit is taken for any salvage value that might be realized from the sale of potential assets during or after decommissioning.

10.1.2.7 Disposal of Waste

All waste produced during decommissioning will be collected, handled, and disposed of in a manner similar to that described for those wastes produced during normal operation. Wastes will consist of normal industrial trash, non-hazardous chemicals and fluids, small amounts of hazardous materials, and radioactive wastes. The radioactive waste will consist primarily of crushed centrifuge rotors, trash, and citric cake. Citric cake consists of uranium and metallic compounds precipitated from citric acid decontamination solutions. This waste may be subject to further volume reduction processes prior to disposal.

Radioactive waste will ultimately be disposed of in licensed low-level radioactive waste disposal facilities. Hazardous waste will be disposed of in hazardous waste disposal facilities. Non-hazardous and non-radioactive waste will be disposed of in a manner consistent with good industrial practice and in accordance with all applicable regulations. A complete estimate of the wastes and effluent to be produced during decommissioning will be provided in the DP that will be submitted prior to initiating the decommissioning of the plant.

For cost estimate purposes, the DUF_6 will be transported at the DOE DUF_6 conversion facility for conversion and disposal.

10.1.2.8 Final Status Survey

A final status survey of the radiological conditions of the Facility is performed to verify proper decontamination. The evaluation of the final status survey is based in part on an initial radiation survey performed prior to initial operation. The initial survey determines the natural background radiation of the area; providing a datum for measurements which determine any increase in levels of radioactivity.

The final status survey will systematically take measurements and perform sampling to describe radioactivity over the entire site. 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 (NRC 2022).

10.1.3 Management and Organization

OE will develop an appropriate organizational strategy to support the phased decommissioning. The organizational strategy will ensure that adequate numbers of experienced and

knowledgeable personnel are available to perform the technical and administrative tasks required to decommission the Facility.

Management of the decommissioning program will ensure that proper training and procedures are implemented to assure worker health and safety. Programs and procedures will focus on minimizing waste volumes and worker exposure to hazardous and radioactive materials. Qualified contractors assisting with decommissioning will likewise be subject to the Facility security and training requirements and procedural controls.

10.1.4 Health and Safety

Consistent with the policy during operation of the Facility, the policy during decommissioning is to keep individual and collective occupational radiation exposures ALARA. The Radiation Protection Program will identify and control sources of radiation, establish worker protection requirements, and direct the use of survey and monitoring instruments.

10.1.5 Waste Management

Radioactive and hazardous wastes produced during decommissioning will be collected, handled, and disposed of in accordance with all regulations applicable to the Facility at the time of decommissioning. Generally, procedures will be similar to those described for wastes produced during normal operation. These wastes will ultimately be disposed of in licensed radioactive or hazardous waste disposal facilities located elsewhere. Non-hazardous and non-radioactive wastes will be disposed of consistent with good industrial practice, and in accordance with applicable regulations.

10.1.6 Security and Nuclear Material Control

Requirements for informational and physical security and for Material Control and Accounting will be maintained during decommissioning in a manner similar to the programs in force during operation. The DP submitted near the end of plant life will provide a description of any necessary revisions to these programs.

10.1.7 Recordkeeping

Records important for safe and effective decommissioning of the Facility will be maintained in accordance with established Records Management and Document Control procedural requirements, and the regulatory requirements of 10 CFR 30.35(g), 10 CFR 40.36(f), and 10 CFR 70.25(g). Information maintained in these records includes:

1. Records of spills or other unusual occurrences involving the spread of contamination in and around the Facility, equipment, or site. These records may be limited 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 nuclides, quantities, forms, and concentrations.
2. As-built drawings and modifications of structures and equipment in restricted areas where radioactive materials are used and/or stored and of locations of possible inaccessible contamination such as buried pipes which may be subject to contamination. Required drawings will be referenced as necessary, although each relevant document will not be

indexed individually. If drawings are not available, appropriate records of available information concerning these areas and locations will be substituted.

3. Except for areas containing only sealed sources, a list contained in a single document and updated every two years of the following:
   i. All areas designated and formerly designated as Restricted Areas as defined under 10 CFR 20.1003;
   ii. All areas outside of Restricted Areas that require documentation specified in item 1 above;
   iii. All areas outside of Restricted Areas where current and previous wastes have been buried as documented under 10 CFR 20.2108; and
   iv. All areas outside of Restricted Areas that contain material such that, if the license expired, OE would be required to either decontaminate the area to meet the criteria for decommissioning in 10 CFR 20, subpart E, or apply for approval for disposal under 10 CFR 20.2002.
4. Records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning, and records of the funding method used for assuring funds if either a funding plan or certification is used.

10.1.8 Decontamination

10.1.8.1 Overview

The methodology, facilities, procedures, and expected results of decontamination are described in the paragraphs below.

The standard decontamination methodology expected to be used during Facility decommissioning will employ conventional decontamination techniques. As described in Section 10.1.2.1 above, the buildings and components are characterized with respect to radioactive contamination immediately prior to the start of decommissioning. The actual decontamination method or methods to be used to decontaminate and decommission the Facility will be confirmed or modified based upon the findings of the site characterization survey. The final decontamination methodology will be described in detail in the DP to be submitted to the NRC prior to commencing decommissioning activities. OE will call upon past experience and lessons learned from previous decommissioning efforts to effectively and efficiently decontaminate the Facility.

10.1.8.2 Equipment

As described in Section 10.1.2.2, new equipment will be installed, and decontamination will be accomplished in existing plant buildings (such as the Centrifuge Assembly Building). The estimated time for equipment installation is approximately one year. These new equipment will be completed in time to support the dismantling and decontamination activities.

The new equipment will be installed into four functional areas that include: (1) a disassembly area, (2) a buffer stock area, (3) a decontamination area, and (4) a scrap storage area for cleaned stock.

Equipment in the decontamination area may include:

- Transport and manipulation equipment
- Dismantling area

- Sawing machines
- Dismantling boxes and tanks (e.g., B-25 boxes)
- Degreasers
- Citric acid and/or other suitable decontamination fluids and demineralized water baths
- Contamination monitors
- Wet blast cabinets
- Crushers or size reduction equipment
- Smelting and/or shredding equipment
- Scrubbing facility.

10.1.8.3 Procedures

Formal procedures for all major decommissioning activities, including decontamination, will be developed and approved by plant management in accordance with the management measures described in Chapter 11. The goal of the procedures will be to minimize worker exposure and waste volumes, and to ensure work is carried out in a safe manner.

10.1.8.4 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 may include process piping, trash, and residue from the effluent treatment systems.

Overall, no problems are anticipated that will prevent the site from being released for unrestricted use.

10.2 DECOMMISSIONING COSTS AND FINANCIAL ASSURANCE

This section provides a general description of decommissioning costs and explains the arrangements made to assure funding is available to cover these costs. A more detailed description of these costs is provided in the DFP.

10.2.1 Facility Decommissioning Cost Estimate

The Table A-20 in the DFP provides a summary of the cost estimate for the decommissioning of the Facility. Costs are provided in FY 2025 dollars with a 25 percent contingency factor added based on the NRC guidance in NUREG-1757 (NRC 2012). Since costs will likely change between the time of license issuance and actual decommissioning, OE will adjust the cost estimate prior to operation of each additional increment of capacity and after full capacity is reached, and no less frequently than every three years consistent with the requirements of 10 CFR 70.25(e). The method for adjusting the cost estimate will consider the following:

- Changes in general inflation (such as, labor rates, consumer price index),
- Changes in price of goods (such as, packing materials),
- Changes in price of services (such as, shipping and disposal costs)
- Changes in facility condition or operations,

- Changes in decommissioning technologies and equipment, and
- Changes in decommissioning procedures or regulations.

The elements of the decommissioning cost estimate are explained below.

10.2.1.1 Planning and Preparation

Scope to include the development of the project execution plan and schedule, development and submittal of the DP, and review and approval of the DP by the NRC.

10.2.1.2 Decontamination and/or Dismantling of Radioactive Facilities

This is based upon utilizing salary and hourly workers at their respective average cost over a five-year duration. Estimated decommissioning costs are based on decontaminating the Facility to the radiological criteria for unrestricted use in 10 CFR 20.1402.

10.2.1.3 Restoration of Contaminated Areas on Facility Grounds

No facility grounds contamination is anticipated because routine radiological surveys will detect contamination and remove it. If an accidental release of radiological material was to occur and the Facility grounds were contaminated to an extent that decontamination during operations is not feasible, the DFP will be updated to include remediation costs to be incurred during final decommissioning.

10.2.1.4 Final Status Survey

This is based upon utilizing salary technicians at their current average cost distribution. Costs do not include any NRC confirmatory surveys to verify the results of the Final Status Survey.

10.2.1.5 Site Stabilization and Long-Term Surveillance

Site stabilization and long-term surveillance (that is, institutional controls) will not be required because the site will be released for unrestricted use. European experience with decommissioning gas centrifuge uranium enrichment plants has been that there is no resultant ground contamination. As a result, site stabilization and long-term surveillance will not be required and associated decommissioning provisions are not provided.

10.2.1.6 Packing, Shipping, and Disposal of Radioactive Waste

This is based upon shipping and disposal of the cascade components, feed and withdrawal equipment, and other components. The packaging cost includes reusable (e.g., rented sealand) and non-reusable (e.g., B-25 boxes) packaging.

10.2.1.7 Equipment and Supplies

This includes the purchase or lease of decontamination equipment and chemicals, small tools and consumables, and supply of utilities.

10.2.1.8 Laboratory

This includes labor costs for sampling, transport, testing, and analysis of samples.

10.2.1.9 Miscellaneous

This includes NRC review and inspection fees, license fees, business insurance, taxes and other period-dependent expenses.

10.2.2 DUF_6 Disposition

DUF_6 are stored in standard cylinders until they can be processed in accordance with the disposal strategy established by OE. Depending on technological developments and the existence of facilities available prior to Facility shutdown, the tails may have commercial value and may be marketable for further enrichment or other processes. However, for the purposes of calculating the DUF_6 disposition cost, OE assumes that the total quantity of tails generated during operation are processed and disposed by the DOE DUF_6 conversion facilities in Portsmouth, Ohio or Paducah, Kentucky. As with Facility decommissioning, the cost estimate will likely change between the time of license issuance and actual decommissioning. OE commits to adjust the cost estimate for DUF_6 disposal annually. The method for adjusting the cost estimate will consider the same factors as previously described in Section 10.2.1 of this chapter. The basis for the DUF_6 disposition estimate is provided in the DFP.

10.2.3 Financial Assurance

Per the exemption request in Chapter 1 of this license application, the financial assurance for decommissioning the plant and disposal of DUF_6 will be provided incrementally as each additional increment of capacity is installed and operated, and DUF_6 are generated. The Facility phased construction and commissioning allows enrichment operations to begin as additional capacity are being constructed and commissioned. Thus, the decommissioning liability is incurred incrementally as additional increment of capacity is installed and operated and DUF_6 production ramps up.

OE is requesting an appropriate exemption to incrementally fund the Facility decommissioning and disposition of DUF_6. In this manner, financial assurance will be available when needed and will be made available as the decommissioning liability is incurred.

Financial assurance will be provided in the form of a surety bond or other guarantee method as required by 10 CFR 30.35(f), 10 CFR 40.36(e), and 10 CFR 70.25(f). Upon finalization of the specific funding instruments to be utilized and prior to the commencement of operations, OE will supplement its application to include the signed, executed documentation.

10.3 REFERENCES

NRC. 2012. "Consolidated Decommissioning Guidance - Financial Assurance, Recordkeeping, and Timeliness." *U.S. Nuclear Regulatory Commission* NUREG-1757 (Volume 3).

NRC. 2022. "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria." *U.S. Nuclear Regulatory Commission* NUREG-1757 (Volume 2).

Chapter 11 – Management Measures

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11.0 MANAGEMENT MEASURES

The management measures adopted by Orano Enrichment USA LLC (OE) to ensure the availability and reliability of items relied on for safety (IROFS) integrate lessons learned from GB2 Enrichment Facility. Management measures described in this application are applied to IROFS or other items or activities designated by OE. IROFS are those items (Configuration Items, CI) or activities of personnel which are relied on to prevent potential accidents at a facility that could exceed the performance requirements in 10 CFR 70.61 or to mitigate their potential consequences. Non-IROFS are those CI or activities of personnel which are not designated as IROFS. Management measures are applied to the IROFS in a graded approach allowing OE to strategically allocate resources while providing reasonable assurance that IROFS are available and reliable to perform their intended function(s) when needed.

OE has elected to classify CI and personnel activities as one of three quality levels (Quality Level 1, Quality Level 2, and Quality Level 3). Quality levels are determined based on the level of risk and safety significance per the methodology outlined in the Integrated Safety Analysis (ISA). Quality level 1 and 2 CI and personnel activities are designated as IROFS due to their contribution to prevent or mitigate potential accidents. Quality Level 3 are those CI and personnel activities that are not designated as IROFS. Management Measures and additional QA elements are outlined in the OE Quality Assurance Program Description (QAPD). These measures are applied to Quality Level 1 and 2 IROFS and other CI or personnel activities as designated commensurate with the complexity of the IROFS, the level of significance, and importance to safety providing reasonable assurance that they are available and reliable to perform their functions when needed.

Management measures are implemented through a quality assurance (QA) program described in the OE's QAPD. The use of the QAPD ensures that in accordance with management measures and other QA elements, that the design, implementation, construction, operation, and decommissioning of IROFS are controlled commensurate with the complexity of the service, the level of significance, and importance to safety.

OE maintains full responsibility for ensuring that the Facility is designed, constructed, tested, and operated in conformance with good engineering practices, applicable regulatory requirements and specified design requirements and in a manner to protect the health and safety of the public. OE and its contractors implement these management measures using approved procedures.

11.1 CONFIGURATION MANAGEMENT

This section discusses oversight, control, and documentation of design and safety information including the changes or modification of IROFS that may impact their ability to be available or reliable to perform their intended function(s) when needed. OE's QAPD, Section 3, Design Control, discusses the design process, design inputs, design verification, configuration management, interface control, and the resulting records.

11.1.1 Configuration Management Program

The Configuration Management Program will ensure that the initial design and safety bases are established, maintained, and documented. This program ensures the integrity of the design and safety bases of the Facility. The different elements of the program will be systematically applied and/or graded to the IROFS ensuring that the IROFS are available and reliable to perform their intended function(s) when needed. Design and safety bases are established using the guidance of the QAPD, Section 3, Design Control. As part of establishing these bases, each of the design documents will be prepared in accordance with approved procedures and undergo

interdisciplinary reviews during the initial release, and subsequent revisions to ensure alignment of design and safety bases. The Configuration Management Program ensures strict consistency among Facility design and operational requirements, the physical configuration, and Facility documentation will be periodically (not to exceed 24 months) assessed for discrepancies between the design and safety bases. This assessment will ensure that changes are appropriately documented, and any corrective and preventative measures are taken. OE maintains the physical configuration of the Facility in strict alignment with design and safety bases through a combination of post-modification testing, periodic assessments/audits/walkdowns, and robust change control procedures described in Section 11.1.4

Deviations to the design bases shall be analyzed by a cross-functional team (typically composed of the design and ISA teams) to determine if there are any impacts to the ISA. Changes or impacts to the ISA are tracked, documented, reviewed, and approved prior to implementing a change. Changes that are approved by the cross-functional team are reviewed to see if NRC approval is needed prior to implementation.

Changes that do not decrease the level of safety do not need to be reviewed by the NRC prior to implementation. However, changes that decrease the level of safety or that are otherwise required to be reviewed by NRC shall be submitted to the NRC for review and approval prior to implementation. All cross-functional team approved changes (regardless of whether they require NRC approval) are documented appropriately reported at least annually to the NRC.

11.1.2 Design Requirements

Design requirements and associated design bases are established and maintained by the Engineering department through a design process that proceeds logically from the design basis to drawings and other technical or design documents. All design documents, including software, are subject to design control requirements. The design process ensures that safety commitments in the ISA are translated into the detailed design and technical documents via a documented engineering workflow, ensuring that final design and technical documents accurately represent the established design basis. Design Control is outlined in the QAPD, Section 3, Design Control. These requirements include personnel training/qualification, control of inputs, process control (e.g., procedures), verification (e.g., independent reviewer, cross functional review), and validation.

11.1.3 Document Control

Procedures are established which control the preparation and issuance of documents such as manuals, instructions, drawings, procedures, specifications, procurement documents and supplier-supplied documents, including any changes thereto. Measures are established to ensure documents, including revisions, are adequately reviewed, approved, and released for use by authorized personnel. Document Control is addressed further in the QAPD, Section 6, Document Control.

11.1.4 Change Control

Procedures control changes to the design and safety bases. The process includes an appropriate level of technical, management, safety, quality, and licensing review and approval prior to implementation. This process includes the conduct of interdisciplinary reviews that constitute a primary mechanism for ensuring consistency of the design with the design and safety basis. Prior to making a change Ci or activities of personnel, the following steps will be taken to evaluate, implement and track the change to ensure compliance with 10 CFR 70.72.

1. Proposed changes will be summarized (documented) and presented to a cross-functional team made up of management, engineering, operations/maintenance, procurement, safety, quality, and licensing. The summary should include the following: the proposed changes, justification (technical basis for the change), temporary changes and duration (if needed), and anticipated impacts (procedural, training or retraining, safety and health, control of licensed material, design and/or safety bases, project schedule, and project cost).
2. Cross-functional team will review and determine if there are any other impacts caused by the change. Prior to implementation, the change shall be internally approved (appropriate authorization) and the change shall be screened to determine if NRC approval is required (see 10 CFR 70.72). Once the appropriate approvals are received, the team evaluates additional actions need to occur to ensure reasonable assurance for the availability and reliability of IROFS. The change and the resultant actions shall be documented in accordance with approved procedures.

The establishment of these controls will ensure the affected IROFS will continue to be available and reliable to perform their intended function(s) when needed, the appropriate parties are notified when required, and that the affected documents are recorded per the appropriate procedures. To prevent inadvertent use of outdated design and technical documents, revisions to affected onsite documentation, including procedures and drawings, are completed and made available to personnel promptly following the implementation of a change

11.1.5 Assessments

As part of establishing a configuration management program, OE will conduct an assessment to ensure the program provides adequate oversight and control of design and safety information, and records of temporary and permanent modifications that may affect the ability of IROFS to be available and reliable to perform their intended safety functions. Both the initial and periodic assessments of the configuration management program are conducted to determine the program's effectiveness and to rectify deficiencies. Periodic assessments include review of the adequacy of documentation and system walk downs of the as-built Facility. Such assessments are conducted and documented in accordance with procedures and scheduled as discussed in the QAPD, Section 18, Audits.

Periodic assessments of the configuration management program and of the design confirm that the system meets its goals and that the design and safety information, and records of modifications are consistent with the design and safety bases. Incident investigations occur in accordance with the QAPD, Section 16, Corrective Action, in the event problems are encountered. Prompt corrective actions are developed as a result of incident investigations or in response to adverse audit or assessment results, in accordance with the QAPD.

11.2 MAINTENANCE

This section outlines the maintenance and functional testing programs to be implemented for the operations phase of the Facility. Preventive maintenance activities, surveillance, and performance trending provide reasonable and continuing assurance that IROFS will be available and reliable to perform their functions.

The purpose of planned and scheduled maintenance for IROFS is to ensure that the equipment and controls are kept in a condition of readiness to perform the planned and designed functions when required. The maintenance group plans, schedules, tracks (documents), and maintains records for maintenance activities. All maintenance (preventative, corrective, functional testing, and surveillance/monitoring) shall be performed and documented in accordance with approved maintenance procedures. The degree at which maintenance will be performed is ascertained as part of the design process, or as a result of monitoring, an audit or assessment finding, incident investigation, corrective action, or a lesson learned.

11.2.1 Surveillance/Monitoring

Surveillance/monitoring is utilized to detect degradation and adverse trends of CI so that action may be taken prior to the CI failure. The monitored parameters are selected based upon their ability to detect the predominant failure modes of the CI. Surveillance/monitoring and reporting are required for CI that are designated as IROFS. IROFS level of surveillance/monitoring like the degree of maintenance is ascertained as part of the design process, and is dependent on the complexity or needs of the service, the level of risk, the level of significance, and importance to safety.

Plant performance criteria are established to monitor plant performance and to monitor IROFS functions and component parameters. These criteria are established and documented using industry experience, operating data, surveillance data, and plant equipment operating experience. These criteria ensure the reliability and availability of IROFS. The performance criteria are also used to demonstrate that the performance or condition of an IROFS is being effectively controlled through appropriate predictive and repetitive maintenance strategies so that they are available and reliable to perform their intended function(s) when needed.

Surveillance of IROFS is performed at specified intervals. The purpose of the surveillance program is to measure the degree to which they meet performance specifications and to determine if the controls applied to the item (administrative or otherwise) are adequate. The results of surveillances are trended, and when the trend indicates potential performance degradation, preventive maintenance frequencies are adjusted, or other appropriate corrective action is taken. As part of the corrective actions, modifications to the preventive maintenance scope and frequency are required to ensure safe and reliable operation of IROFS.

OE verifies that administrative controls identified as IROFS are available and reliable through periodic procedural audits, assessments, training proficiency evaluations, and performance monitoring as part of the surveillance function to ensure they are available and reliable during extended periods of operation.

Incident investigations may identify root causes of failures that are related to the type or frequency of maintenance. The lessons learned from investigations are incorporated into the surveillance/monitoring and preventive maintenance programs as appropriate.

Maintenance procedures prescribe compensatory measures, if appropriate, for surveillance tests of IROFS that can be performed only while equipment is out of service.

Records showing the current surveillance schedule, performance criteria, and test results for all IROFS will be maintained and retained in accordance with the record management system.

Results of surveillance/monitoring activities related to IROFS via the configuration management program will be evaluated by all safety disciplines to determine if there are any impacts on the ISA.

11.2.2 Corrective Maintenance

Corrective maintenance involves repair or replacement of equipment that has unexpectedly degraded or failed. Corrective maintenance of IROFS restores the equipment to acceptable performance through a planned, systematic, controlled, and documented approach for the repair and replacement activities. As part of the investigation, scheduled maintenance activities and their frequency will be evaluated for adequacy or to see if maintenance or lack thereof was the cause or contributing cause of premature degradation or failure.

Following corrective maintenance on an IROFS, and before returning it to operational status, functional testing, if necessary, is performed to ensure the IROFS is available and reliable to perform their intended function(s) when needed.

Results of corrective maintenance activities related to IROFS via the configuration management program will be evaluated by applicable safety disciplines to determine any impact on the ISA, and any updates needed.

Corrective maintenance activities involving the replacement of IROFS components are conducted as 'like-for-like' replacements whenever possible, maintaining the original design and safety basis. In instances where a 'like-for-like' replacement is not practicable, the proposed replacement is treated as a Facility change and evaluated and screened per 10 CFR 70.72 as described in Section 11.1.4. This ensures that any modification to the physical configuration resulting from corrective maintenance is systematically evaluated for its impact on the Integrated Safety Analysis (ISA) and that the IROFS remains available and reliable to perform its intended safety function before being returned to operational status.

11.2.3 Preventive Maintenance

Preventive maintenance (PM) includes preplanned and scheduled periodic refurbishment, partial or complete overhaul, or replacement of IROFS, if necessary, to ensure their continued safe operation. Planning for preventive maintenance includes consideration of results of surveillance and monitoring, including failure history. PM also includes instrument calibration and testing.

The PM program procedures and calibration standards (traceable to the national standards system or to nationally accepted calibration techniques, as appropriate) enable the Facility personnel to calibrate equipment and monitoring devices important to plant safety and safeguards. Testing performed on IROFS that are not redundant will provide for compensatory measures to be put into place to ensure that the IROFS function is performed until it is put back into service. Graded maintenance for IROFS may include extended calibration intervals, provided that such intervals do not exceed manufacturer recommendations or accepted industry standards and are justified by established performance data.

Industry experience, operating data, surveillance data, and plant equipment operating experience are used to determine initial PM frequencies and procedures. In determining the frequency of PM, consideration is given to appropriately balancing the objective of preventing failures through maintenance against the objective of minimizing unavailability of IROFS because of PM. In addition, feedback from PM and corrective maintenance and the results of incident investigations

and identified root causes are used, as appropriate, to modify the frequency or scope of PM. The rationale for deviations from industry standards or vendor recommendations for PM is documented.

Results of preventive maintenance activities related to IROFS via the configuration management system will be evaluated by all safety disciplines to determine impact on the ISA and any updates needed.

11.2.4 Functional Testing

Functional testing of IROFS is performed appropriately, following initial installation, as part of periodic surveillance testing, and after corrective or preventive maintenance or calibration to ensure that the IROFS meets its original design requirements and safety parameters defined in the ISA and is available and reliable to perform their intended function(s) when needed.

Results of surveillance/monitoring activities related to IROFS via the configuration management program will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

Testing (preoperational, operational, maintenance required, periodic, and special) of IROFS is performed and controlled in accordance with the QAPD, Section 11, Test Control. Each of the testing modes: preoperational, operational, maintenance required, periodic, and special testing are conducted to ensure safe and reliable operation of the Facility. For new processes or during Facility startup, functional tests are designed to include all operational aspects of the IROFS intended safety functions. Records showing the functional test schedule and results for all IROFS maintenance are maintained and retained in accordance with the records management system.

If testing is performed during operation and an IROFS is out of service, localized outages will be utilized to ensure the impacted IROFS are isolated. Lock Out Tag Out and testing procedures may be utilized ensure that personnel are aware of localized outage and which equipment is unavailable. Localized outages and other compensatory measures will ensure continued safety operation.

11.3 TRAINING AND QUALIFICATIONS

The training program requirements apply to those plant personnel who perform activities that are classified as IROFS. Each type of training shall be conducted in accordance with reviewed and approved training materials (guides, courses, lesson plans, guides, etc.). Training materials shall clearly document the purpose of the training, training/learning objectives, relevant details (such as job performance requirements), standards of performance, and standards for evaluating acceptable trainee performance.

The principal objective of the OE training program is to ensure job proficiency of all Facility personnel involved in IROFS activities through effective training and qualification. The training program is designed to accommodate future growth and meet commitments to comply with applicable established regulations and standards. In some circumstances, there may be the need to grant exemptions from training, these exemptions may be granted when the need is justified, documented and approved by the relevant management.

Qualification is indicated by successful completion of prescribed training, demonstration of the ability to perform assigned tasks and where required by regulation, maintaining a current and valid certification. Training is designed, developed and implemented according to a systematic approach. Employees are provided with formal training to establish the foundation of knowledge and on-the-job training to develop work performance skills. Continuing training is provided, as

required, to maintain proficiency in these knowledge and skill components, and to provide further employee development.

OE's Training program is organized under the direction of the Operations Manager. Line management is responsible for the content and conduct of training. OE utilizes performance-based training as the primary management tool for developing, conducting and evaluating training. Training and qualification documentation, including exemptions are documented and recorded and retained for traceability.

11.3.1 Position Training Requirements

Each discipline shall have their own curated training modules or guides. These modules/guides provide the trainee(s) with an understanding of the work within their purview. Minimum qualification requirements are developed and included in Chapter 2 of this License Application for those who perform regulated activities and other personnel as required. Initial identification of job-specific training requirements is based on experience. The objective of the training shall be to ensure safe and efficient operation of the Facility and compliance with applicable established regulations and requirements. Training requirements shall be applicable to, but not necessarily restricted to, those personnel within the Operations Group who have a direct relationship with the operation, maintenance, testing or other technical aspect of IROFS. Training courses are kept up to date to reflect plant modifications and changes to procedures when applicable. Relevant trainings are linked in the configuration management system to ensure that the training accurately reflects Facility modifications and design changes in a timely manner.

Periodic retraining courses shall be established when applicable to ensure that personnel remain proficient. Periodic retraining generally is conducted to ensure retention of knowledge and skills important to Facility operations. The training may consist of periodic retraining exercises, instruction, and review of subjects as appropriate to maintain proficiency of all personnel assigned to the Facility

11.3.2 Bases and Objectives for Training

The training program incorporates insights from the ISA and Human Factors Engineering (HFE). Objectives are developed to address error-likely situations and human to system interfaces for IROFS, ensuring that personnel qualifications account for the cognitive and physical demands identified during the safety analysis process.

Learning objectives identify the training content, as established by needs/job analyses and position-specific requirements. The task list from the needs/job analysis is used to develop action statements that describe the desired post-training performance. Objectives include the knowledge, skills, and abilities the trainee should demonstrate; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

11.3.3 Training Guides

Training guides are developed and maintained under the direction of the Operations Manager. These guides ensure that trainees have a good understanding of the task and can perform tasks competently. Training guides are records that are controlled per the QAPD.

11.3.4 Training Evaluations

Trainee understanding and command of learning objectives is evaluated through observation/demonstration or oral/written tests as appropriate. Such evaluations measure the trainee's skills and knowledge of job performance requirements.

Qualified personnel shall be evaluated to identify the training programs strengths and weaknesses. OE uses feedback from trainee performance during training and former trainees and their supervisors to evaluate and refine training.

Training evaluations is one of the mechanisms that may be used to “retrain” or “requalify” personnel. Alternatively, periodic or repeat training or qualification may be used ensure personnel are comfortable and competent in their job duties.

11.3.5 On-the-Job Training

On-the-job training in conjunction with classroom training (or assigned reading) shall be used for tasks that require a higher level of control or for IROFS that require or need specific on-the-job training. Designated personnel, competent in the program standards and methods of conducting the training, conduct on-the-job training using current performance-based training materials. Completion of on-the-job training is demonstrated by actual task performance or performance of a simulation of the task with the trainee explaining task actions using the conditions encountered during the performance of the task, including references, tools, and equipment reflecting the actual task to the extent practical.

11.3.6 Training Program Assessments

The Training Program shall be assessed annually per QAPD, Section 18, Audits. The assessment shall ensure that personnel are properly trained, certified, qualified, and competent in performing their assigned activities. A tertiary benefit of the assessments is to check the need for retraining or reevaluations for qualifications. If the assessment discovers a shortcoming, the shortcoming will be documented (utilizing the corrective action program as necessary) and remedial steps will be taken to ensure that impacted IROFS are still available to reliably perform their intended functions.

The annual assessments will also ensure alignment with the design and safety bases. Change to either the design or safety basis will result in an investigation to determine if training is impacted as part of the Configuration Management Program. Additionally, the assessments will verify that training materials remain consistent with the HFE assumptions used in the ISA and that any identified human performance trends are incorporated into the continuing training cycle.

11.3.7 Personnel Qualification

The QAPD, Section 2, Quality Assurance Program provides training and qualification requirements, for personnel performing IROFS work activities; for nondestructive examination, inspection, and test personnel; and for QA auditors. Other personnel that are required to be qualified for work activities shall conduct the applicable training as outlined above.

11.4 PROCEDURES DEVELOPMENT AND IMPLEMENTATION

Procedures are prepared, maintained, and implemented per OE's QAPD, Sections 5, Instructions, Procedures and Drawings, and 6, Document Control. OE shall have procedures that protect the public's health and safety. Measures including the Configuration Management Program and training program shall ensure that the latest revision of procedures is available and utilized.

11.4.1 Procedures

Procedures are governed by a "master" procedure which establishes methodology for the preparation, review, approval, revision, cancellation and control for policies, procedures, plans, guides, manuals, and forms. Procedures will be reviewed prior to release and periodically reviewed to ensure that they can be performed as written, technically accurate, and aligned with the QAPD and industry's best practices.

Procedures shall be developed based on the needs of the organization. The ISA, QAPD, management measures, operations, etc., should be utilized to determine if a procedure may be needed. Prior to implementation, procedures should be validated using mechanisms such as field testing.

Procedures should include the following as applicable:

- Title, document number, revision and date
- Purpose and scope (limitations) of the procedure
- References
- Responsibilities (who is responsible and accountable for the actions)
- Background (including prerequisites and precautions)
    - Actions for normal and off-normal conditions of operation
- Procedural steps (including human actions and limitations)
- Acceptance criteria
- Check lists
- Operating limits and IROFS
- Safety checkpoints

11.4.2 QA and Administrative Procedures

To develop a QA Program, procedures will be developed for the relevant QA elements, Management Measures, construction (as appliable), radiation safety, criticality safety, fire safety, chemical process safety, etc. These implementing procedures will support the QAPD.

Facility administrative procedures are written by each department as necessary to control activities that support process operations, including management measures. Listed below are several areas for which administrative procedures are written:

A. Operator's authority and responsibility

B. Activities affecting Facility operation or operating indications

C. Manipulation of Facility control

D. Relief of Duties

E. Equipment control

F. Master surveillance testing schedule

G. Control Room Operations

H. Fire Protection Procedures

I. Abnormal Events

Administrative procedures and controls are determined as part of the ISA evaluation and may be augmented or changed due to the corrective action program, lessons learned, or updates to the design or safety basis (communicated via the Configuration Management Program). These procedures and controls will be frequently evaluated for effectiveness as part of internal audits and assessments. The audits and amendments will evaluate the administrative procedures and controls to ensure they support the IROFS being reliable and available when needed. The application of these procedures and controls is supported by the training program to ensure that the controls and procedures are up to date and accurately implemented.

Activities involving licensed materials and/or IROFS will be conducted in accordance with approved procedures. These procedures are intended to provide a pre-planned method of conducting operations of systems to eliminate errors due to on-the-spot analysis and judgments. These procedures like administrative procedures are reviewed to ensure that they are aligned with the QAPD, industry best practices, and to ensure they capture lessons learned. By using industry best practices and incorporating lessons learned, the use of procedures will eliminate or mitigate the risk associated with these activities.

11.4.3 Changes to Procedures

Temporary procedures are subject to the same level of review and approval as permanent procedures. The use of temporary procedures is limited to a documented timeframe, and they are issued only when permanent procedures do not exist to direct operations during testing, maintenance, and modifications, or to provide guidance in unusual situations. Temporary changes to procedures shall not involve a change to the ISA and shall not alter the intent of the original procedure.

11.4.4 Maintenance and Testing Procedures

Maintenance of Facility CI is performed in accordance with written procedures, documented instructions, checklists, or drawings appropriate to the circumstances that conform to applicable codes, standards, specifications, and other appropriate criteria. Maintenance procedures should include steps for a pre-maintenance review/inspection, required stop work notifications, impact to surrounding CI, maintenance personnel requirements, configuration management requirements, testing requirements, documentation requirements, safe work requirements etc.

Testing conducted on a periodic basis to determine various Facility parameters and to verify the continuing capability of IROFS to meet performance requirements is conducted in accordance with approved, written procedures. Periodic test procedures are utilized to perform such testing and are sufficiently detailed that qualified personnel can perform the required functions without direct supervision. Testing performed on IROFS that are not redundant will provide for compensatory measures to be put into place to ensure that the IROFS performs until it is put back into service.

Procedures will include provisions for operations to stop and place the process in a safe condition if a step of a procedure cannot be performed as written. In the event of unexpected incidents (such as an accident, unexpected transient, significant operator error, or equipment malfunction), procedures should be reviewed to determine if a modification is needed.

Maintenance procedures describe the controls for the procurement and use of replacement components to ensure like-kind replacement and adherence to 10 CFR 21 requirements for IROFS.

11.5 AUDITS AND ASSESSMENTS

OE utilizes a tiered approach to verifying compliance with procedures and the effectiveness of management measures. The system of audits and assessments is designed to provide comprehensive program oversight and ensure that IROFS are available and reliable to perform their intended safety functions. Audits, assessments and the application graded or otherwise are described OE's QAPD, Section 18 Audits.

11.5.1 Audit and Assessment Program

Audits and assessments are conducted in the areas of radiation safety, nuclear criticality safety, chemical safety, fire safety, environmental protection, emergency management, QA, configuration management, maintenance, training and qualification, procedures, incident investigation, and records management.

Audits are performed to verify that operations are being conducted in accordance with regulatory requirements and license commitments. Assessments are focused on the effectiveness of activities and ensuring that management measures are achieving their intended purpose.

11.5.2 Independence and Qualifications

Qualified personnel without direct responsibility for the function and area being audited or assessed perform the audits and assessments. To ensure objective oversight, independent assessments are conducted by offsite groups or individuals not involved in the licensed activity to verify that the health, safety, and environmental compliance functions are effectively achieving their intended purposes.

11.5.3 Results and Corrective Action

Audit and assessment results are documented and reported to the Company President, senior management, and relevant safety committees. The results are used to influence lessons learned which are communicated to the workforce driving future improvements. Findings and recommendations are tracked, and corrective actions are implemented in accordance with the corrective action program to rectify deficiencies and mitigate or eliminate recurrence.

11.5.4 Graded Approach to Oversight

Audit and assessment activities are applied in a graded manner based on safety significance:

Frequency: QL-1 IROFS and high-risk activities are subject to more frequent oversight. QL-2 IROFS may be subject to reduced audit and assessment frequencies.

Personnel: Assessments associated with graded IROFS may be performed by personnel knowledgeable of the specific trade or skill involved in the activity in lieu of personnel with specific training in the conduct of formal assessments.

11.6 INCIDENT INVESTIGATIONS AND CORRECTIVE ACTION PROGRAM

The incident investigation process is a simple mechanism available for use by any person at the Facility for reporting deficiencies, abnormal events and potentially unsafe conditions or activities. Abnormal events that potentially threaten or lessen the effectiveness of health, safety or environmental protection will be identified and reported to and investigated by the EHS&L Manager. The investigation determines the specific or generic root cause(s), generic implications, and risk significance of the event to facilitate the application of lessons learned. Each event will be considered in terms of its requirements for reporting in accordance with regulations and will be evaluated to determine the level of investigation required. Potential defects and non-compliances shall be evaluated in accordance with 10 CFR 21 to determine if a potential substantial safety hazard exists. These investigations including the process of incident identification, root cause analysis, environmental protection analysis, recording, reporting, and follow-up are addressed in and performed by the Corrective Action Program (CAP) procedures. Radiological, criticality, hazardous chemical, and industrial safety requirements are addressed. Guidance for classifying occurrences are contained in CAP procedures, including examples of threshold off-normal occurrences. The depth of the investigation will depend upon the severity of the classified incident in terms of the levels of uranium released and/or the degree of potential for exposure of workers, the public or the environment. Investigations include a comparison of the event sequence with accident sequences already considered in the ISA. The ISA summary is modified per approved procedures to include evaluation of the risk associated with accidents of the type actually experienced to ensure safety significance is correctly characterized. Lessons learned from these investigations are proactively disseminated to relevant Facility personnel through the training program, updated procedures, or safety bulletins to prevent recurrence and enhance the overall safety culture

The discovery of an event that prevents immediate protective actions necessary to avoid exposures to radiation or radioactive materials that could exceed regulatory limits or releases of licensed material that could exceed regulatory limits (events may include fires, explosions, toxic gas releases, etc.) or incidents involving licensed material will be evaluated immediately (as soon as practical) by the EHS&L Manager, reporting individual, and the relevant stake holders. This immediate evaluation will ensure that “1 hour” or “4 hour” notifications can be made within a timely matter. Other issues will be evaluated to ensure compliance with requirements notification requirements. If a notification is deemed necessary, the EHS&L Manager shall provide notification to the NRC within the required timeline.

The OE QAPD identifies the responsibilities and provides authority for those individuals involved in quality activities to identify any condition adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective materials and equipment, and non-conformances. These individuals identify and document conditions adverse to quality, analyze and determine how the conditions can be corrected or resolved, and take such steps as necessary to implement corrective actions in accordance with documented procedures. The rigor of the incident investigation process, including the size and composition of the investigation team, is commensurate with the risk significance of the incident and the safety significance of the involved IROFS.

The QAPD requires regularly scheduled audits and assessments to ensure that the corrective actions needed are identified. OE employees have the authority and responsibility to initiate corrective actions if they discover deficiencies. The QAPD contains procedures for identifying, reporting, resolving, documenting, and analyzing conditions adverse to quality. Reports of conditions adverse to quality are analyzed to identify trends in quality performance. Significant

conditions adverse to quality and significant trends are reported to senior management in accordance with CAP procedures.

Conditions adverse to quality or failure of an IROFS or CI, the cause of the conditions, and the corrective action taken to mitigate repetition shall be documented, recorded, and reported to management for review and assessment in accordance with CAP procedures. Documentation related to abnormal events is maintained for the life of the operation.

The QAPD, Section 16, Corrective Action, provides additional details regarding the CAP requirements.

11.7 RECORDS MANAGEMENT

Records management is performed by the Records Management organization. Records management procedures define the responsibilities and specify the authority required for record retention or disposal, ensuring that access is controlled and the system remains effective through periodic assessments. The organization ensures that records are managed in a controlled and systematic manner to provide legible, identifiable and retrievable documentation. Records are categorized by quality level and subject matter to identify record protection needs and to designate retention periods. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures. QA records are not considered valid until they are authenticated and dated by authorized personnel.

The QA Program contains procedures for preparing, reviewing, approving, handling (transmitting, distributing, protecting etc.), identifying/characterizing, retaining, retrieving and maintaining quality assurance records. Procedures ensure that records are protected against tampering, theft, loss, unauthorized access, damage, or deterioration while in storage. These records are maintained at locations where they can be reviewed and audited to establish that the required quality has been assured.

IROFS performance including their failures shall be documented, maintained, and revised as necessary to document follow-up actions and investigation results. Failures shall be recorded at the time of discovery. Record revisions necessitated by post-failure investigations are made promptly after completion of the investigation in accordance with 10 CFR 70.62(a)(3).

For computer codes and computerized data used for IROFS, procedures are established for maintaining readability and usability of older codes and data as computing technology changes.

When a single records storage Facility is used, it is reviewed for adequacy of protecting the records by a person competent in the technical field of fire protection and fire extinguishing. Dual records storage facilities are not subject to this review.

Records will be retained to ensure conformance with 10 CFR Part 70. The table below highlights the minimum retention period for each type of document. OE will ensure that these records retention requirements meet or exceed these requirements. In addition to being retained, records will be legible, easily traceable, categorized by safety importance, and linked back to specific IROFS.

Record Type	Minimum Retention Period
IROFS	Life of the facility
Safety Program Records (ISA and IROFS failures etc.)	Life of the facility
Decommissioning	Until license termination
Material Control and Accounting Changes	5 years after each change
Visitor Registers and Access Logs	3 years after last entry
Accidental Criticality Reports	Until license termination
Alarm and Security Incident Logs	3 years after each change
Waste Disposal Records	Until license termination
Superseded Security Procedures	3 years after each change
Effluent Monitoring Report	Until license termination

QAPD, Section 17, Quality Assurance Records, of this chapter provides additional details regarding records management requirements. As with other sections of the QAPD and Management Measures, periodic assessments will be conducted to promptly detect and correct deficiencies in the records management system or in its implementation.

11.8 OTHER QA ELEMENTS

The QAPD establishes 18 QA elements which are applied to management measures, IROFS, and other designated CI and personnel activities. The application of the QA elements is performed in a graded manner. The selection of quality levels is commensurate with the risk involved and specifically parallels the risk informed grading established for maintenance, training, and surveillance needed to ensure a consistent, integrated safety approach across the Facility life cycle.

The QA elements described in the table below and in the QAPD are applied to provide reasonable assurance that IROFS will be designed, procured, and maintained satisfying the requirements of 10 CFR 70.61 for the protection of workers, the public, and the environment.

QAPD Section Number	Section Title	NUREG-1520 / 10 CFR 70 Management Measure	Primary Purpose in 10 CFR 70 Context
1	Organization	11.3 (Training & Qualifications) / Chapter 2 (Org)	Independence of QA from Production
2	Quality Assurance Program	11.0 (Management Measures)	Establishes the Graded Approach (QL-1, 2, 3)
3	Design Control	11.1 (Configuration Management)	Maintains the Design Basis and 10 CFR 70.72 controls
4	Procurement Document Control	11.1 (Configuration Management) / 11.8 (Other QA)	Flow down of IROFS & 10 CFR 21 requirements

QAPD Section Number	Section Title	NUREG-1520 / 10 CFR 70 Management Measure	Primary Purpose in 10 CFR 70 Context
5	Instructions, Procedures, and Drawings	11.4 (Procedures)	Ensuring work is performed via approved methods
6	Document Control	11.1 (Configuration Management) / 11.4 (Procedures)	Controlling revisions of the design and safety basis
7	Control of Purchased Items and Services	11.2 (Maintenance) / 11.8 (Other QA)	Acceptance of Items
8	Identification and Control of Items	11.1 (Configuration Management) / 11.2 (Maintenance)	Traceability of IROFS from receipt to install
9	Control of Special Processes	11.2 (Maintenance)	Qualification of IROFS welding/NDE
10	Inspection	11.2 (Maintenance)	Verification of IROFS physical condition
11	Test Control	11.2 (Maintenance)	Functional Testing post-maintenance/install
12	Control of Measuring and Test Equipment	11.2 (Maintenance)	Accuracy of IROFS calibration & monitoring
13	Handling, Storage, and Shipping	11.2 (Maintenance)	Preserving IROFS reliability before use
14	Inspection, Test, and Operating status	11.2 (Maintenance)	Preventing use of unverified/failed IROFS
15	Control of Nonconforming Items	11.6 (Incident Investigation/CAP)	Controlling IROFS that fail to meet spec
16	Corrective Action	11.6 (Incident Investigation/CAP)	10 CFR 21 Evaluations & Root Cause Evaluations
17	Quality Assurance Records	11.7 (Records Management)	Documentary evidence of IROFS reliability
18	Audits	11.5 (Audits & Assessments)	Periodic verification of program effectiveness

Chapter 12 – MC&A

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12.0 MATERIAL CONTROL AND ACCOUNTING

Pursuant to the requirements in 10 CFR Part 70, the License Application is seeking a license for Category III Enrichment Facility (i.e., producing enrichments below 10 wt%^{235}U).

Following the requirements in 10 CFR 70.22(b) and 74.33(b), a description of the uranium enrichment facility program for control and accounting of the Nuclear Material in its possession under license is provided in the Fundamental Nuclear Material Control Plan (FNMCP).

Chapter 13 – Physical Protection

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13.0 PHYSICAL PROTECTION

This chapter follows the guidance from NUREG-1520, rev. 2, Ch. 13 for the license application for a Category III Enrichment Facility (i.e., producing enrichments below 10 $^{w}/_{o}$ ^{235}U).

Following the requirements in 10 CFR 70.22(k), 73.67(a), 73.67(c)(1), 73.67(d), 73.67(e), 73.67(f), and 73.67(g), a full description of the Facility physical protection system is provided in the OE Physical Security Plan.