

Enclosure 1
Page changes to the Safety Analysis Report showing changes made under
Revision 25
(Revision bars, strikethroughs and underlines utilized)

SAFETY ANALYSIS REPORT

Revision 25

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Summary of Changes for Revision 24		
Issue/ Date	Change	Description of Change
	LBDCR-10-0022 01-15-10	Material Quantities table being revised to reflect License conditions 6,7 & 8 LAR-09-29
24c continued	LBDCR-10-0004 11-14-10	Relocation of the CUB roof staircase (Figure 1.1-14) CC-EG-2009-0499; 70.72 = 2010-0008
	LBDCR-10-0009 02-25-10	SBM-1001 extension and CAB extension CC-LS-2010-0005; 70.72 = 2010-0190
	LBDCR-10-0036 03-02-10	Autoclave 471=4B1 will not be installed in the SBM 1001 due to space and budget considerations CC-EG-2009-0429; 70.72 = 2010-0188
24d 03-25-10	LBDCR-10-0039 03-09-10	Temporary storage of SBM condensate (in lieu of LECTS) CC-EG-2010-0005; 70.72 = 2010-0016
	LBDCR-10-0042 03-10-10	Refine & clarify terminology used to describe radiological areas. Also update standards and process systems, removing no longer used and adding new. CC-RP-2010-0001; 70.72 = 2010-0222
	LBDCR-10-0037 03-14-10	Distinguish between systems bounded at enrichment of 6 ^w / ₁₀ ²³⁵ U and 1.5 ^w / ₁₀ ²³⁵ U CC-LS-2010-0012; 70.72 = 2010-0189
	LBDCR-10-0044 03-11-10	Operate While Constructing LAR-09-14
	LBDCR-10-0033 03-18-10	Phased Operation is being revised to clearly identify the scope of the individual phases. CC-LS-2010-0010; 70.72 = 2010-0186
	LBDCR-10-0043 03-23-10	Ensure compensatory measures are in place when an IROFS has been compromised. CC-LS-2010-0013; 70.72 = 2010-0228

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Summary of Changes for Revision 24		
Issue/ Date	Change	Description of Change
24d (Continued)	LBDCR-10-0046 03-20-10	Mobile pump and trap set used as local exhaust ventilation for connection of on-line mass spectrometer. (Section has been deleted in prior LBDCR-10-0033) CC-EG-2010-0112; 70.72 = 2010-0247
25 03-25-10	N/A	Submittal to NRC for non substantial changes previously approved by LES

ACRONYMS AND ABBREVIATIONS

NMSLO	New Mexico State Land Office
NMSS	Nuclear Material Safety and Safeguards
NMWQB	New Mexico Water Quality Bureau
NMWQCC	New Mexico Quality Control Commission
NNE	north-northeast
NNW	north-northwest
No.	number
NOAA	National Oceanic and Atmospheric Administration
NOI	Notice of Intent
NPDES	National Pollutant Discharge Elimination System
NPDWS	National Primary Drinking Water Standard
NRC	United States Nuclear Regulatory Commission
NRHP	National Register of Historic Places
NSDWS	National Secondary Drinking Water Standard
NSPS	New Source Performance Standards
NSR	New Source Review
NTS	Nevada Test Site
NWS	National Weather Service
NW	northwest
OEPA	Ohio Environmental Protection Agency
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
OVEC	Ohio Valley Electric Corporation
P&IDs	pipng and instrumentation diagrams
p.	page
PA	public address
<u>PCM</u>	<u>Personnel Contamination Monitor</u>
PEL	Permissible Exposure Level
PFPE	perfluorinated polyether
PGA	peak ground acceleration
pH	measure of the acidity or alkalinity
PHA	Process Hazard Analysis
Ph.D.	Doctor of Philosophy

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ACRONYMS AND ABBREVIATIONS

PIA	Potentially Impacted Area
PLC	Programmable Logic Controllers
PM	preventive maintenance
PM _{2.5}	particulates $\leq 2.5\mu\text{m}$
PM ₁₀	particulates $\leq 10\mu\text{m}$
PMF	probable maximum flood
PMP	Probable Maximum Precipitation
PMWP	Probable Maximum Winter Precipitation
PORTS	Portsmouth Gaseous Diffusion Plant
POTW	Publicly Owned Treatment Works
pp.	pages
PRC	Peoples Republic of China
PSAR	Preliminary Safety Analysis Report
PSP	Physical Security Plan
QA	quality assurance
QAPD	Quality Assurance Program Description
QC	Quality Control
RCB	Radiation Control Bureau
RCRA	Resource Conservation and Recovery Act
<u>RCA</u>	<u>Radiologically Controlled Area</u>
RCZ	radiation control zone
REIS	Regional Economic Information System
REMP	Radiological Environmental Monitoring Program.
RIMS	Regional Input-Output Modeling System
ROI	Region of Interest or Radius of Influence
RTE	Rare Threatened and Endangered
RWP	radiation work permit
S	south
SAR	Safety Analysis Report
SB	Separations Building
Sc.D.	Doctor of Science
SCRAM	Support Center for Regulatory Air Models
SDWA	Safe Drinking Water Act

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1.1 Facility and Process Description

1.1.1 Facility Location, Site Layout, And Surrounding Characteristics

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 220 ha (543 acres) of land in Section 32 of Lea County, New Mexico. The Separations Building Modules, Administration Building, Cylinder Receipt and Dispatch Building, Centrifuge Assembly Building, Central Utilities Building, Technical Services Building, and UBC Storage Pad are located approximately in the center of the Section. A Plot Plan of the facility is shown in Figure 1.1-3, Plot Plan (1 Mile Radius). The Facility Layout (Site Plan) depicting the Site Boundary and Controlled Area Boundary is shown in Figure 1.1-4, Facility Layout (Site Plan) with Site Boundary and Controlled Access Area Boundary.

The site lies along the north side of New Mexico Highway 234. It is relatively flat with slight undulations in elevation ranging from 1,033 to 1,061 m (3,390 to 3,430 ft) above mean sea level (msl). The overall slope direction is to the southwest. During the construction phase, a fence runs along the perimeter of the property. A 254-mm (10-in) diameter, underground carbon dioxide pipeline owned by Trinity Pipeline LLC, traverses the site from southeast to northwest. A 406-mm (16-in) diameter, underground natural gas pipeline, owned by the Sid Richardson Energy Services Company, is located along the south property line, paralleling New Mexico Highway 234.

The nearest community is Eunice, approximately 8 km (5 mi) from the site. There are no residences, schools, stores or other population centers within a 1.6 km (1 mi) radius of the site.

Additional details of proximity to nearby populations are provided in the Environmental Report.

1.1.2 Facilities Description

~~(See 12.1)~~ The major structures and areas of the facility are outlined below.

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Separations Building Modules

~~(See 12.1.1.1)~~ The overall layout of a Separations Building Module (SBM) is presented in Figures 1.1-5 through 1.1-7 and the UF₆ Handling Area is shown in Figure 1.1-8, UF₆ Handling Area Equipment Location. Each SBM consists of two Cascade Halls, each having multiple cascades with each cascade having many of centrifuges. The major functional areas of the Separations Building Modules are:

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- Cascade Halls (2)
- Process Services Corridor
- UF₆ Handling Area

Source material and special nuclear material (SNM) are used or produced in this area.

Technical Services Building

~~(See 12.1.1.2)~~ The overall layout of the Technical Services Building (TSB) is presented in Figures 1.1-9, Technical Services Building First Floor, and 1.1-10, Technical Services Building

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Second Floor. The TSB contains support areas for the facility. It also acts as the secure point of entry to the Separations Building Modules and the Cylinder Receipt and Dispatch Building (CRDB). The major functional areas of the TSB are:

- Environmental Monitoring Laboratory
- Medical Room
- Break Room
- Control Room
- Emergency Operations Center
- Training Room
- Central Alarm Station (CAS)

The Security Diesel Generator provides backup 480 volt power to selected security and security related equipment during a loss of normal power. The Security Diesel Generator is not a requirement for safe operation of the plant. The Security Diesel Generator is designed for outdoor use and will be located south of the TSB. The fuel oil storage tank is sized for 24 hours of continuous operation at 100 percent rated power output.

Centrifuge Assembly Building

This building is used to assemble centrifuges before they are moved into the Separations Building and installed in the cascades. The overall layout of the Centrifuge Assembly Building (CAB) is presented in Figures 1.1-11 and 1.1-12. The Centrifuge Assembly Building is located adjacent to the Cylinder Receipt and Dispatch Building. The major functional areas of the CAB are:

- Centrifuge Component Storage Area
- Centrifuge Assembly Area
- Assembled Centrifuge Storage Area
- Centrifuge Test Facility
- Centrifuge Post Mortem Facility

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

Administration Building

(See 12.1.1.68) The general office areas are located in the Administration Building. Personnel enter the Administration Building and general office areas via the main lobby.

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Security Building

(See 12.1.1.740) The main site Security Building is located at the entrance to the plant. It functions as a security checkpoint for incoming and outgoing personnel. Employees and visitors that have access approval are screened at this location.

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The Security Building also contains a Visitor Center. There are adequate physical barriers, locked doors, etc. to separate the visitor accessible areas from areas designed to support security functions.

A smaller Gatehouse has been placed at the secondary site entrance. Common carriers, such as mail delivery trucks, are screened at this location.

The Entrance Exit Control Point (EECP) is located in the Main Security Building. All personnel access to the facility occurs at this location. Vehicular traffic passes through a security checkpoint before being allowed to park. Parking is located outside of the Controlled Access Area (CAA) security fence. Personnel enter the Security Building area via the main lobby. Personnel requiring access to the facility areas or the CAA must pass through the EECP. The EECP is designed to facilitate and control the passage of authorized facility personnel and visitors.

Entry to the facility area from the Security Building is only possible through the EECP.

Cylinder Receipt and Dispatch Building

(See 12.1.1.3) The overall layout of the Cylinder Receipt and Dispatch Building (CRDB) is presented in Figure 1.1-13, Cylinder Receipt and Dispatch Building First Floor. The CRDB is located between two Separations Building Modules, north of the Technical Services Building. This building contains equipment to receive, inspect, weigh and temporarily store cylinders of feed UF₆ sent to the plant; temporarily store, inspect, weigh, and ship cylinders of enriched UF₆ to facility customers; receive, inspect, weigh, and temporarily store clean empty product and UBCs prior to being filled in the Separations Building; and inspect, weigh, and transfer filled UBCs to the UBC Storage Pad. The CRDB also contains various laboratories and maintenance facilities necessary to safely operate and maintain the facility.

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The functions of the Cylinder Receipt and Dispatch Building are:

Outside the Cylinder Receipt and Dispatch Building's Bunkered Area:

- Loading and unloading of cylinders
- Inventory weighing
- Preparation and storage of protective cylinder overpacks
- Storage of clean empty and empty UBCs
- Buffer storage of feed cylinders
- Semi-finished product storage
- Final product storage
- Prepared cylinder storage

Inside the Cylinder Receipt and Dispatch Building's Bunkered Area:

- Equipment decontamination
- Rebuilding of vacuum pumps
- UF₆ cylinder valve repair

1.1 Facility and Process Description

- Solid waste collection and packaging
- Collection and treatment of liquid effluents
- Contaminated material handling
- Mass spectrometry and chemical analysis
- Radiation monitoring
- Filtration and exhaust of gaseous effluent through Gaseous Effluent Vent Systems (GEVS)
- HVAC (supporting radiological and non-radiological portions of the CRDB)

Source and SNM are used in the above CRDB areas listed above.

UBC Storage Pad

(See 12.1.1.46) The facility utilizes an area outside of the CRDB, the UBC Storage Pad, for storage of cylinders containing UF_6 that is depleted in ^{235}U . The cylinder contents are stored under vacuum in corrosion-resistant ANSI N14.1 Model 48Y cylinders.

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The UBC storage area layout is designed for moving the cylinders with a transporter/mover (e.g., a semi-tractor trailer) and a crane. A transporter/mover moves the UBCs from the CRDB to the UBC Storage Pad entrance. A double girder gantry crane removes the cylinders from the transporter/mover and places them in the UBC Storage Pad. The gantry crane is designed to double stack the cylinders in the storage area.

Source material is used in this area.

Central Utilities Building

(See 12.1.1.57) The Central Utilities Building (CUB) is shown on Figure 1.1-14, Central Utilities Building First Floor. The Central Utilities Building houses two diesel generators, which provide the site with standby power. The rooms housing the diesel generators are constructed independent of each other with adequate provisions made for maintenance, equipment removal and equipment replacement. The building also contains Electrical Rooms/Areas, an Air Compressor Area, and Centrifuge Cooling Water System.

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1.1.3 Process Descriptions

This section provides a description of the various processes analyzed as part of the Integrated Safety Analysis. A brief overview of the entire enrichment process is provided followed by an overview of each major process system.

1.1.3.1 Process Overview

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 uranium hexafluoride (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

1.1 Facility and Process Description

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 UF₆.

1.1.3.2 Process System Descriptions

An overview of the four enrichment process systems and the two 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, Chemical Process Safety, contains a discussion of the criteria and identification of the chemicals of concern at the NEF and concludes that uranium hexafluoride (UF₆) is the only chemical of concern that will be used at the facility. Chapter 6, Chemical Process Safety, also identifies the locations where UF₆ is stored or used in the facility and includes a detailed discussion and description of the hazardous characteristics of UF₆ as well as a detailed listing of other chemicals that are in use at the facility.

The enrichment process is comprised of the following major systems:

UF₆ Feed System

(See ~~12.2.212.1.1.1.3~~ and 12.1.2.1) The first step in the process is the receipt of the feed cylinders and preparation to feed the UF₆ through the enrichment process.

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Natural UF₆ feed is received at the NEF in 48Ycylinders from a conversion plant. Pressure in the feed cylinders is below atmospheric (vacuum) and the UF₆ is in solid form.

The function of the UF₆ Feed System is to provide a continuous supply of gaseous UF₆ from the feed cylinders to the cascades. There are five¹ Solid Feed Stations per Cascade Hall. The maximum feed flow rate is based on a maximum capacity of 545,000 SWU per year per Cascade Hall.

Cascade System

(See ~~12.2.312.1.1.1.1~~ and 12.1.2.2) The function of the Cascade System is to receive gaseous UF₆ from the UF₆ Feed System and enrich the ²³⁵U isotope in the UF₆ to a maximum of 5 %.

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Multiple gas centrifuges make up arrays called cascades. The cascades separate gaseous UF₆ feed with a natural uranium isotopic concentration into two process flow streams – product and tails. The product stream is the enriched UF₆ stream, from 2 - 5 % ²³⁵U, with an average of 4.5 % ²³⁵U. The tails stream is UF₆ that has been depleted of ²³⁵U isotope to 0.20 - 0.34 % ²³⁵U, with an average of 0.32 % ²³⁵U.

Product Take-off System

(See ~~12.2.412.1.1.1.4~~ and 12.1.2.3) The function of the Product Take-off System is to provide continuous withdrawal of the enriched gaseous UF₆ product from the cascades and to purge and dispose of light gas impurities from the enrichment process.

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The product streams leaving the cascades are brought together into one common manifold from the Cascade Hall. The product stream is transported via a train of vacuum pumps to Product LTTS in the UF₆ Handling Area. There are five Product LTTS per Cascade Hall.

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The Product Take-off System also contains a system to purge light gases (typically air and HF) from the enrichment process. This system consists of UF₆ Cold Traps which capture UF₆ 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 carbon trap, an alumina trap, and a vacuum pump. The carbon trap removes small traces of UF₆ and the alumina trap removes any HF from the product gas.

Tails Take-off System

(See ~~12.2-512.1.1.1.4~~ and 12.1.2.4) The primary function of the Tails Take-off System is to provide continuous withdrawal of the gaseous UF₆ tails from the cascades. A secondary function of this system is to provide a means for removal of UF₆ from the centrifuge cascades under abnormal conditions.

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The tails stream exits each Cascade Hall via a primary header, goes through a pumping train, and then to Tails LTTS in the UF₆ Handling Area. There are eight Tails LTTS per Cascade Hall. In addition to the four primary systems listed above, there are two major support systems:

Product Blending System

(See ~~12.2-612.1.1.1.5~~ and 12.1.2.5) The primary function of the Product Blending System is to provide a means to fill 30B cylinders with UF₆ at a specific enrichment of ²³⁵U to meet customer requirements. This is accomplished by blending (mixing) UF₆ at two different enrichment levels to one specific enrichment level. The system can also be used to transfer product from a 30B cylinder to another 30B cylinder without blending.

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

Product Liquid Sampling System

(See ~~12.2-712.1.1.1.6~~ and 12.1.2.6) The function of the Product Liquid Sampling System is to obtain an assay sample from filled product 30B cylinders. The sample is used to validate the exact enrichment level of UF₆ in the filled product cylinders before the cylinders are sent to the fuel processor.

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The Product Liquid Sampling System is one of two systems at NEF that changes solid UF₆ to liquid UF₆. The Sub-Sampling System also changes solid UF₆ to liquid UF₆.

1.1.4 Raw Materials, By-Products, Wastes, And Finished Products

The facility handles Special Nuclear Material of ²³⁵U contained in uranium enriched above natural but less than or equal to 5.0 % in the ²³⁵U isotope. The ²³⁵U is in the form of uranium hexafluoride (UF₆). The facility processes approximately 690 feed cylinders (Model 48Y), 350 product cylinders (Model 30B), and 625 UBCs (Model 48Y) per year.

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

Solid Waste Management

1.1 Facility and Process Description

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(See ~~12.1.3 A~~, 12.1.1.3.1 and 12.1.3.3) Solid waste generated at the NEF will be grouped into industrial (non-hazardous), radioactive, hazardous, and mixed waste categories. In addition, solid radioactive and mixed waste is further segregated according to the quantity of liquid that is not readily separable from the solid material. The solid waste management systems are comprised of a set of facilities, administrative procedures, and practices that provide for the collection, temporary storage, processing, and transportation for disposal of categorized solid waste in accordance with regulatory requirements. All solid radioactive wastes generated are Class A low-level wastes (LLW) as defined in 10 CFR 61 (CFR, 2003a).

Radioactive waste is collected in labeled containers in each Radiation Area and transferred to the Solid Waste Collection Room for processing. Suitable waste will be volume-reduced, and all radioactive waste will be disposed of at a licensed LLW disposal facility.

Hazardous waste and a small amount of mixed waste are generated at the NEF. These wastes are also collected at the point of generation and transferred to the Solid Waste Collection Room. Any mixed waste that may be processed to meet land disposal requirements may be treated in its original collection container and shipped as LLW for disposal.

Industrial waste, including miscellaneous trash, filters, resins and paper is shipped offsite for compaction and then sent to a licensed waste landfill.

Effluent Systems

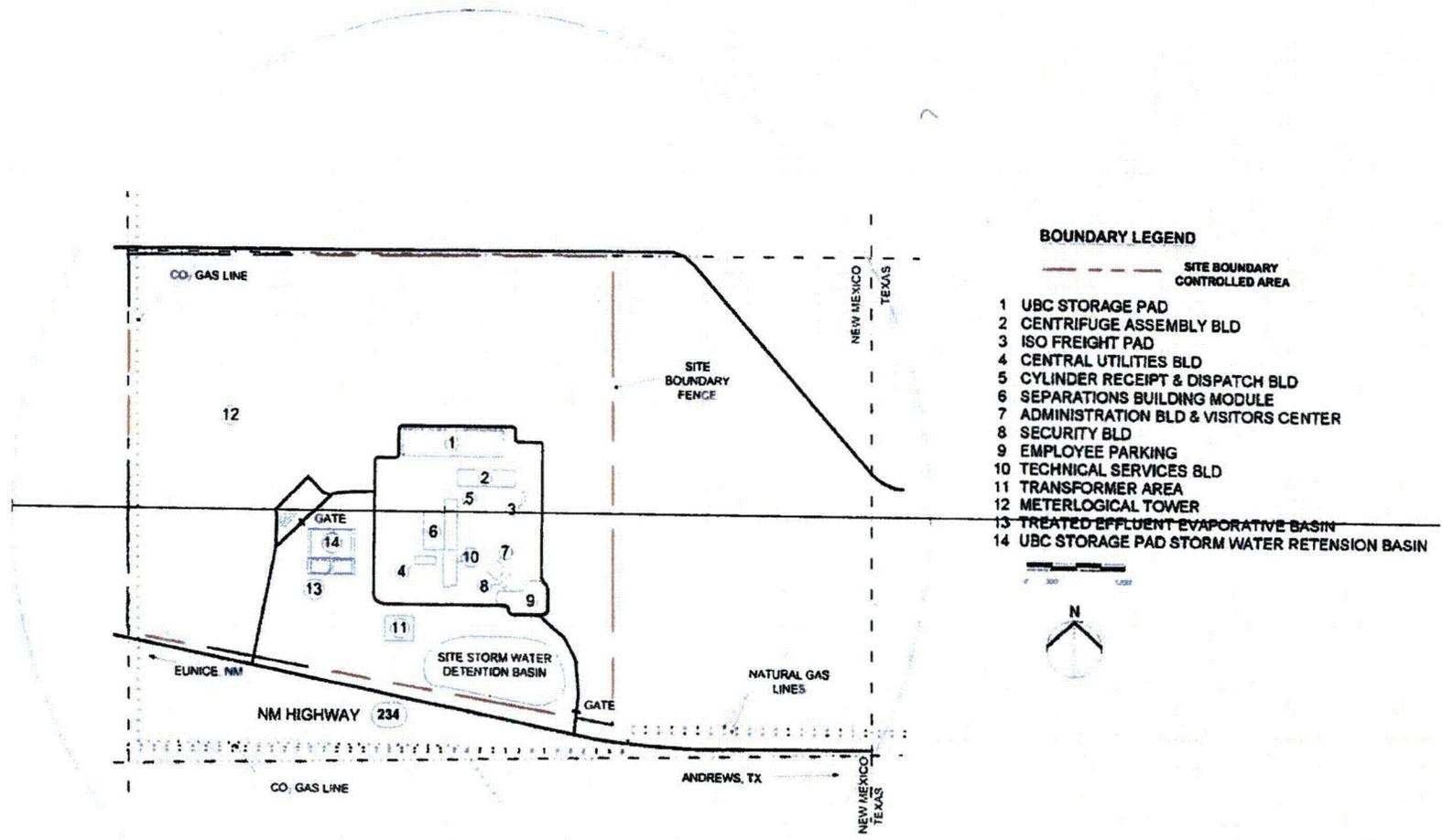
The following NEF systems handle wastes and effluent.

- Pumped Extract GEVS
- CRDB GEVS
- Confinement Ventilation function of CRDB HVAC System
- Liquid Effluent Collection and Treatment System
- Centrifuge Test and Post Mortem Facilities Exhaust Filtration System
- Sewage System
- Solid Waste Collection System
- Decontamination System
- PFPE Oil Recovery System

Effluent Quantities

Quantities of radioactive and non-radioactive wastes and effluent are estimated and shown in the tables referenced in this section. The tables include quantities and average uranium concentrations. Portions of the waste considered hazardous or mixed are identified. The following tables address plant effluents:

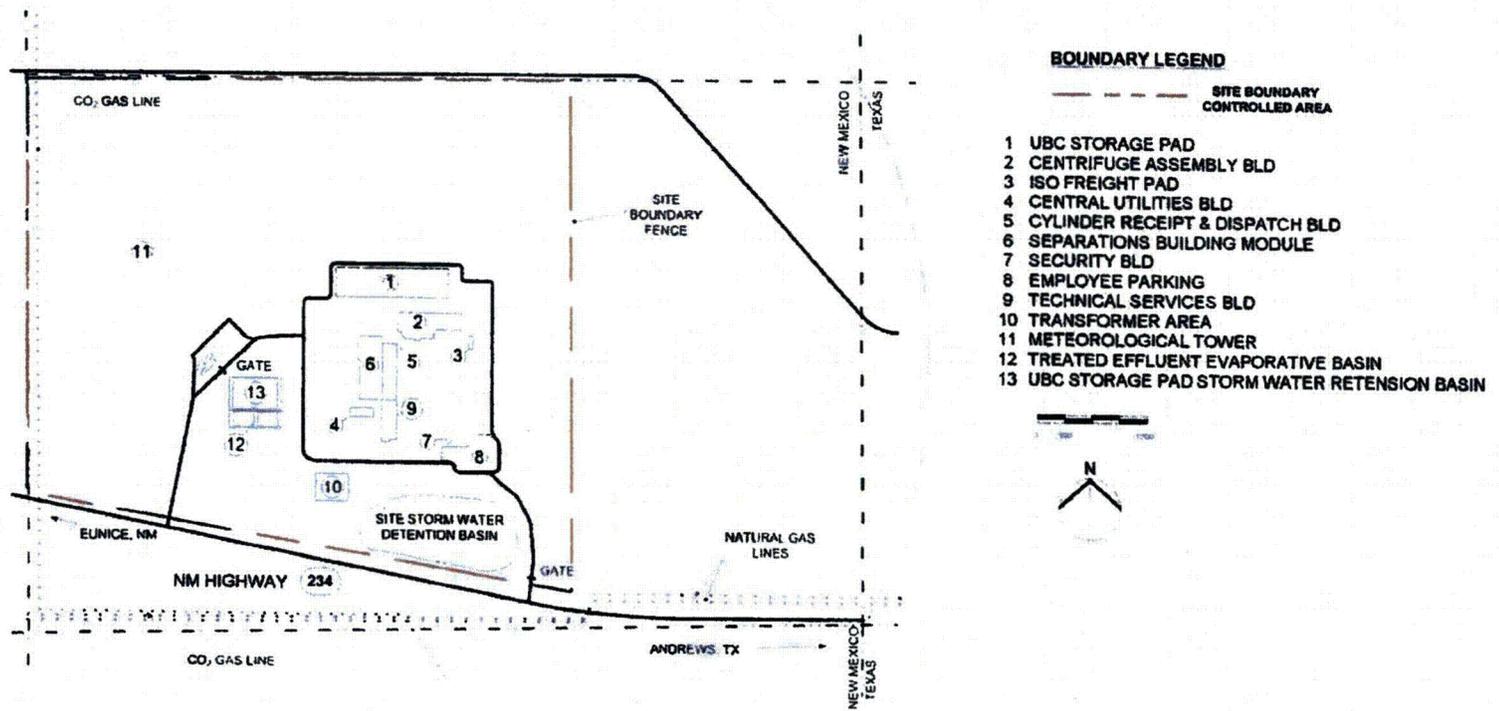
- Table 1.1-1, Estimated Annual Gaseous Effluent
- Table 1.1-2, Estimated Annual Radiological and Mixed Wastes
- Table 1.1-3, Estimated Annual Liquid Effluent



BOUNDARY LEGEND

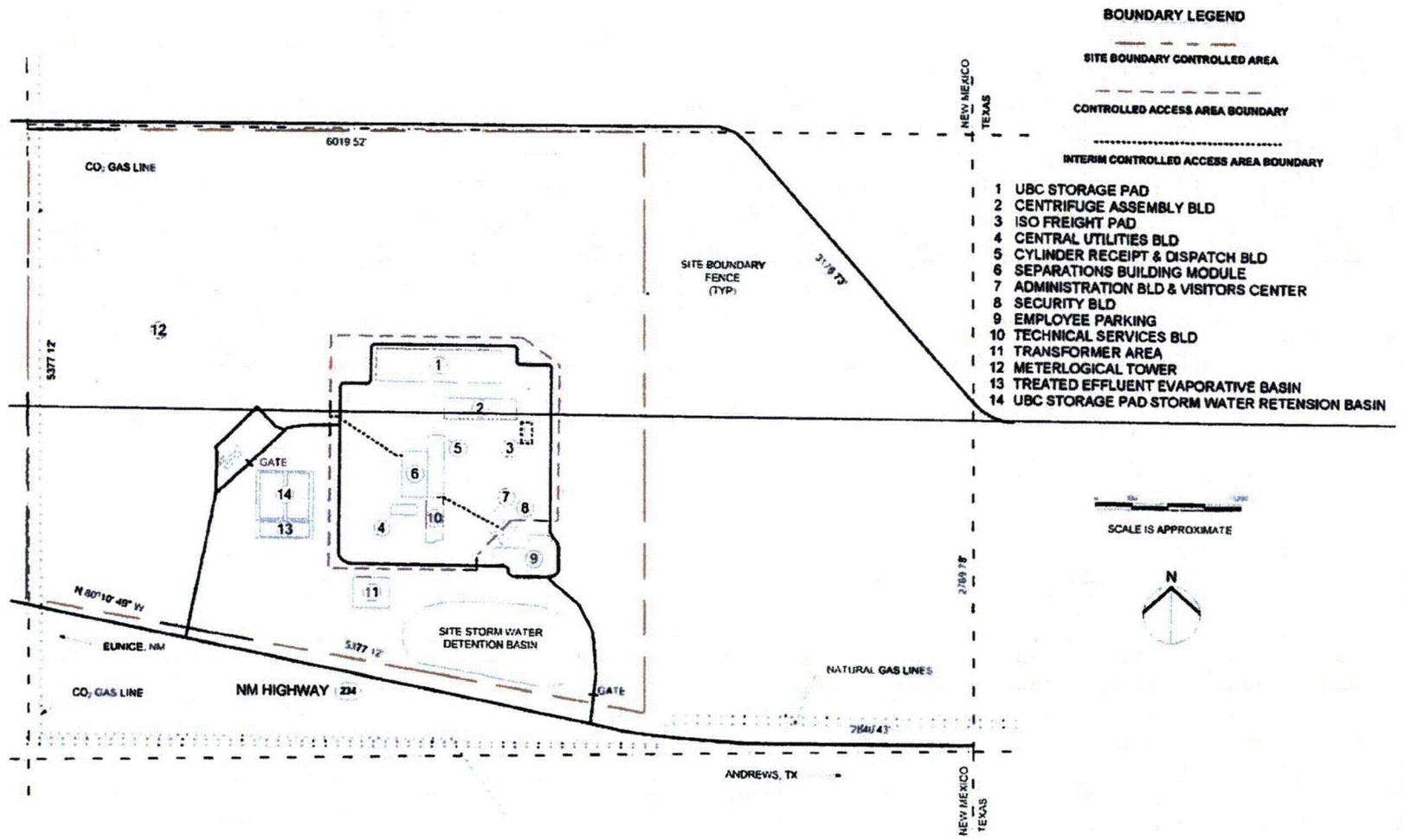
- SITE BOUNDARY CONTROLLED AREA
- 1 UBC STORAGE PAD
- 2 CENTRIFUGE ASSEMBLY BLD
- 3 ISO FREIGHT PAD
- 4 CENTRAL UTILITIES BLD
- 5 CYLINDER RECEIPT & DISPATCH BLD
- 6 SEPARATIONS BUILDING MODULE
- 7 ADMINISTRATION BLD & VISITORS CENTER
- 8 SECURITY BLD
- 9 EMPLOYEE PARKING
- 10 TECHNICAL SERVICES BLD
- 11 TRANSFORMER AREA
- 12 METERLOGICAL TOWER
- 13 TREATED EFFLUENT EVAPORATIVE BASIN
- 14 UBC STORAGE PAD STORM WATER RETENTION BASIN

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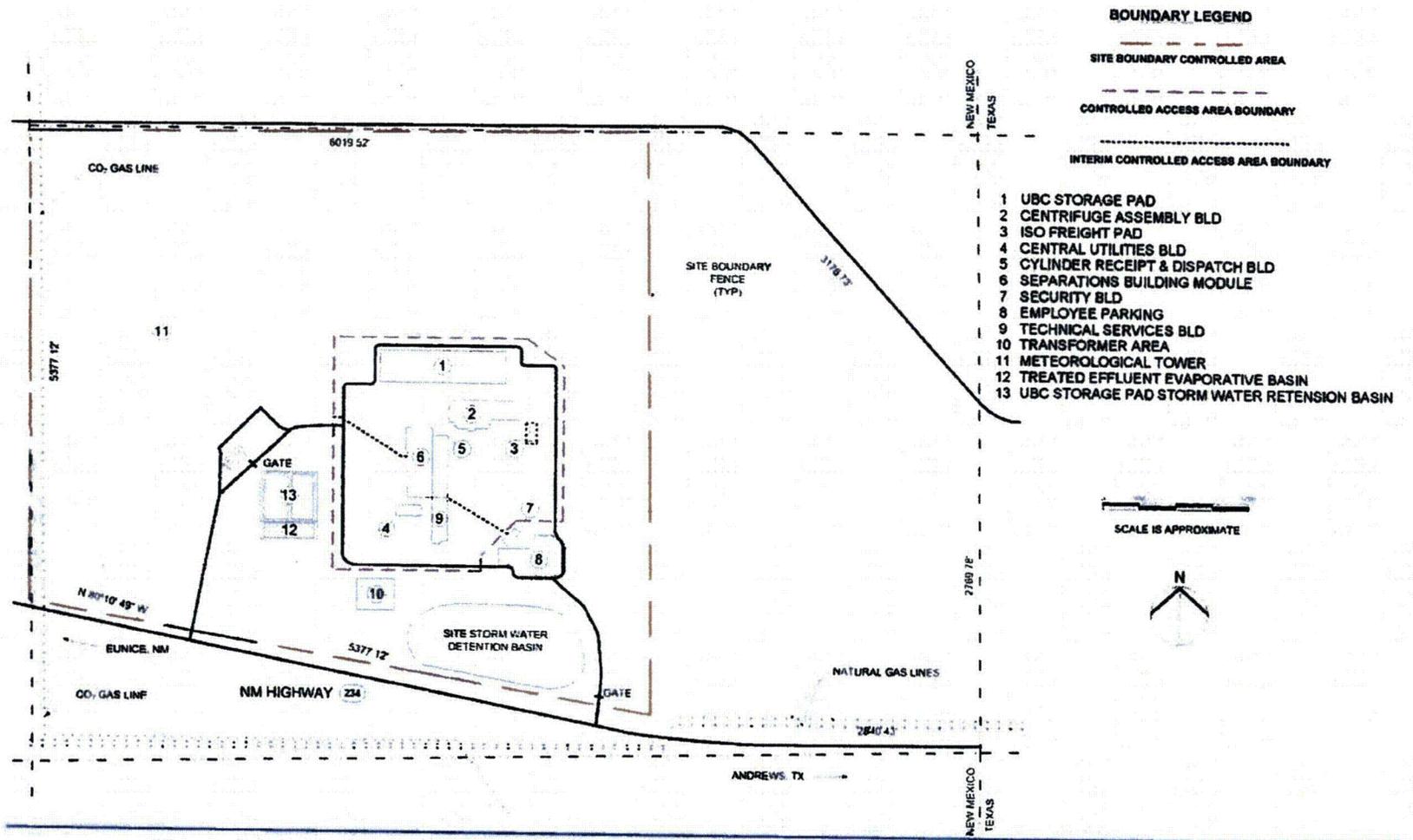
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Figure 1.1-3 Plot Plan (1 Mile Radius)

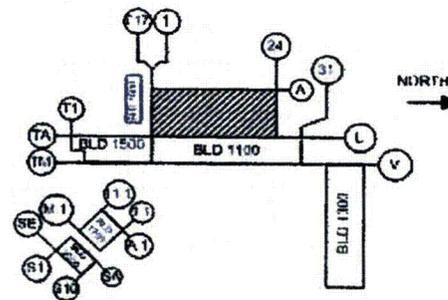
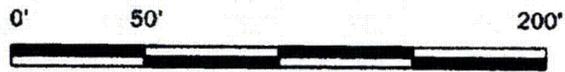
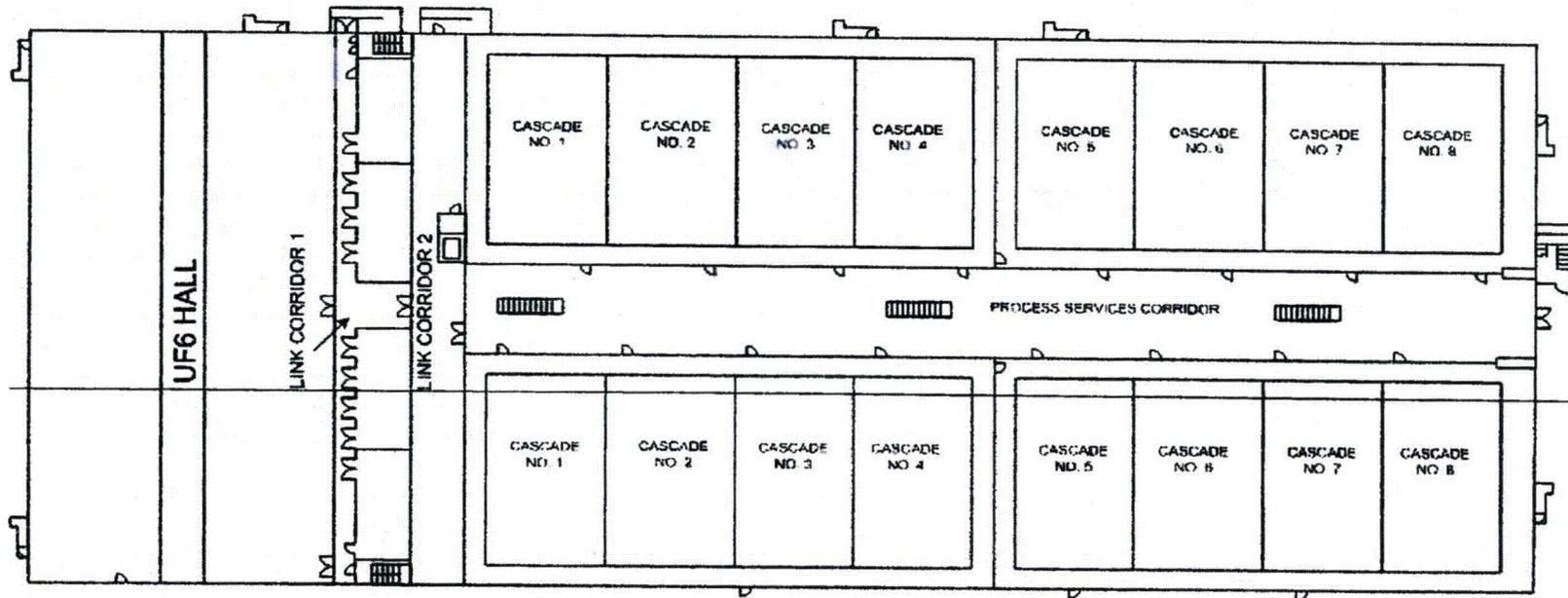


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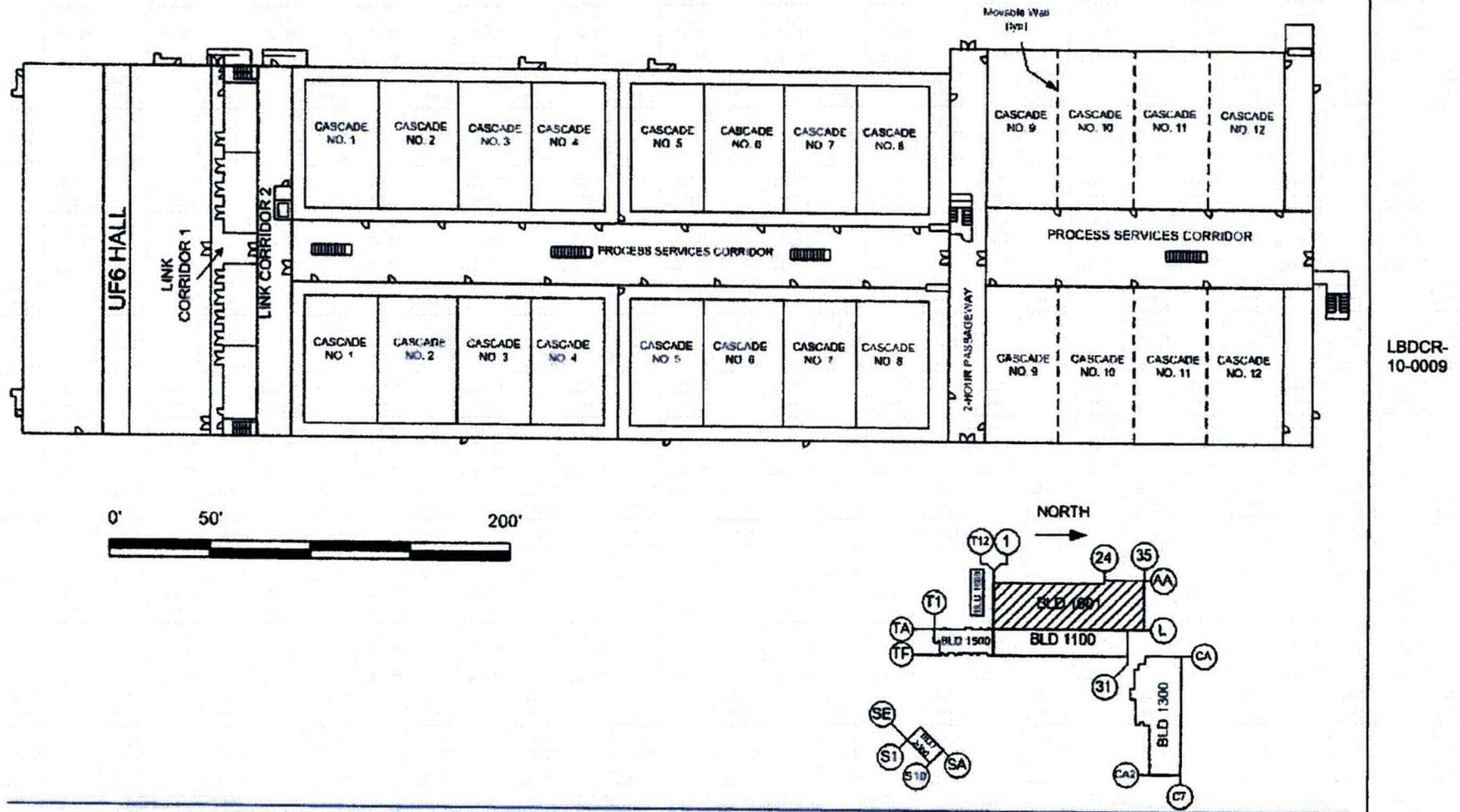
1.6 Chapter 1 Figures



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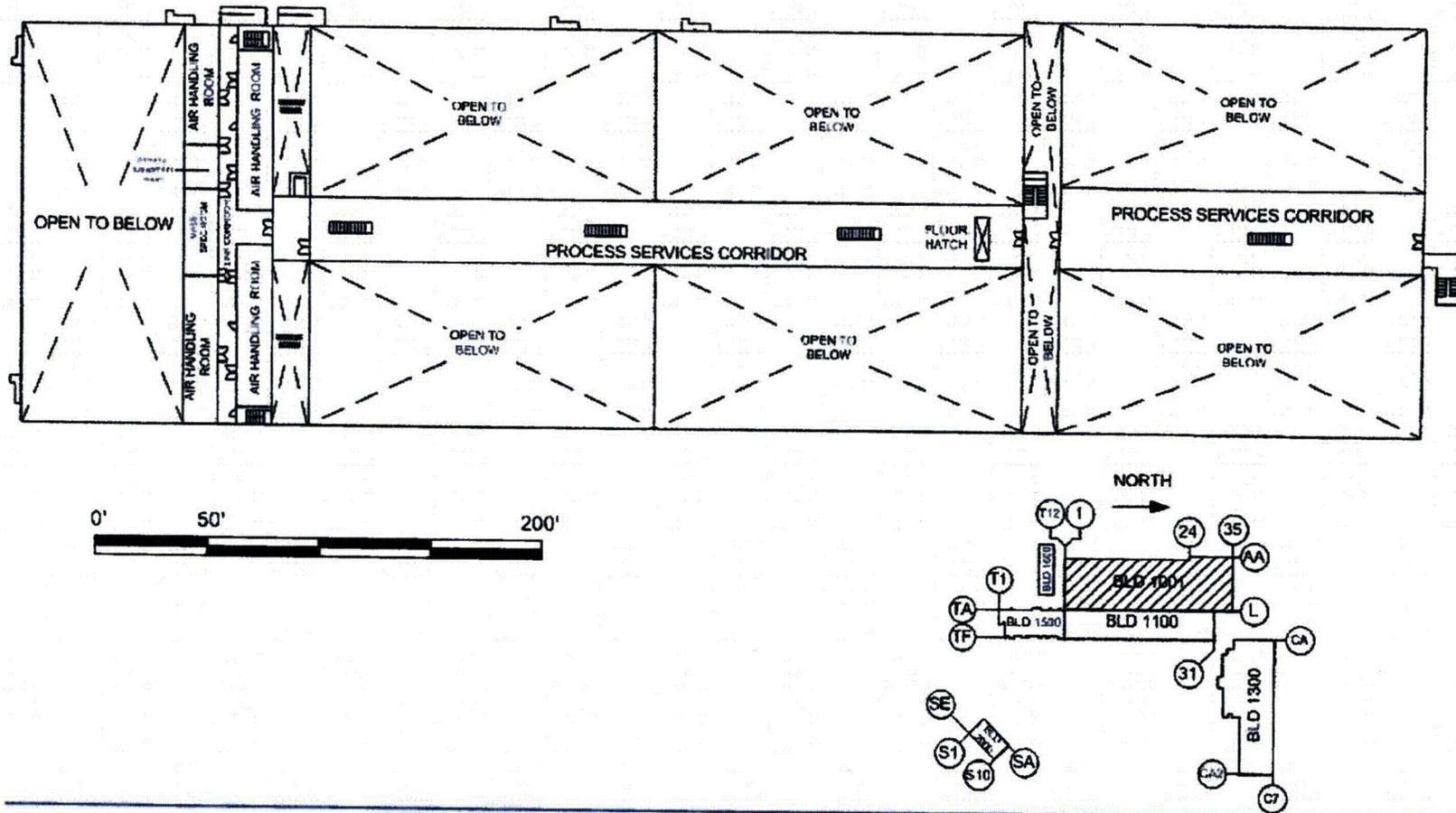


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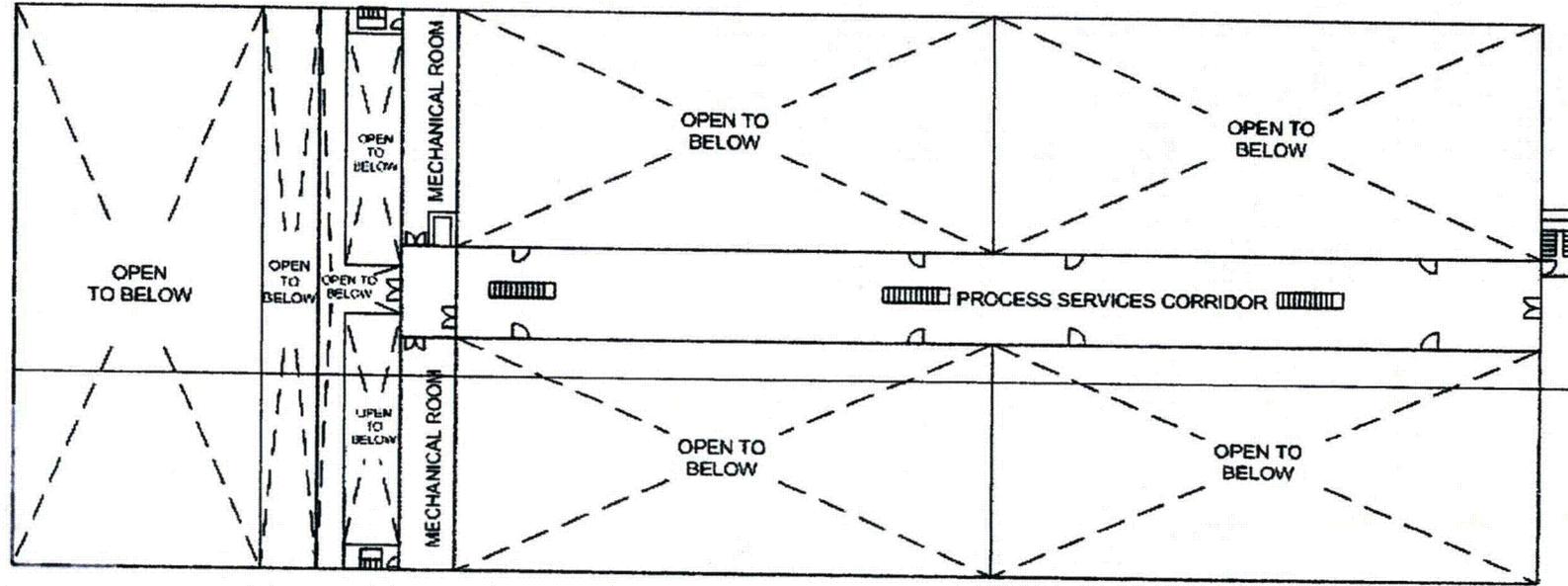
Figure 1.1-5 Separations Building Module First Floor



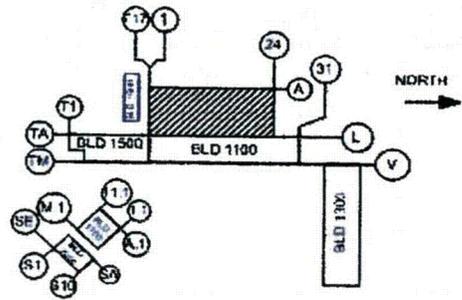
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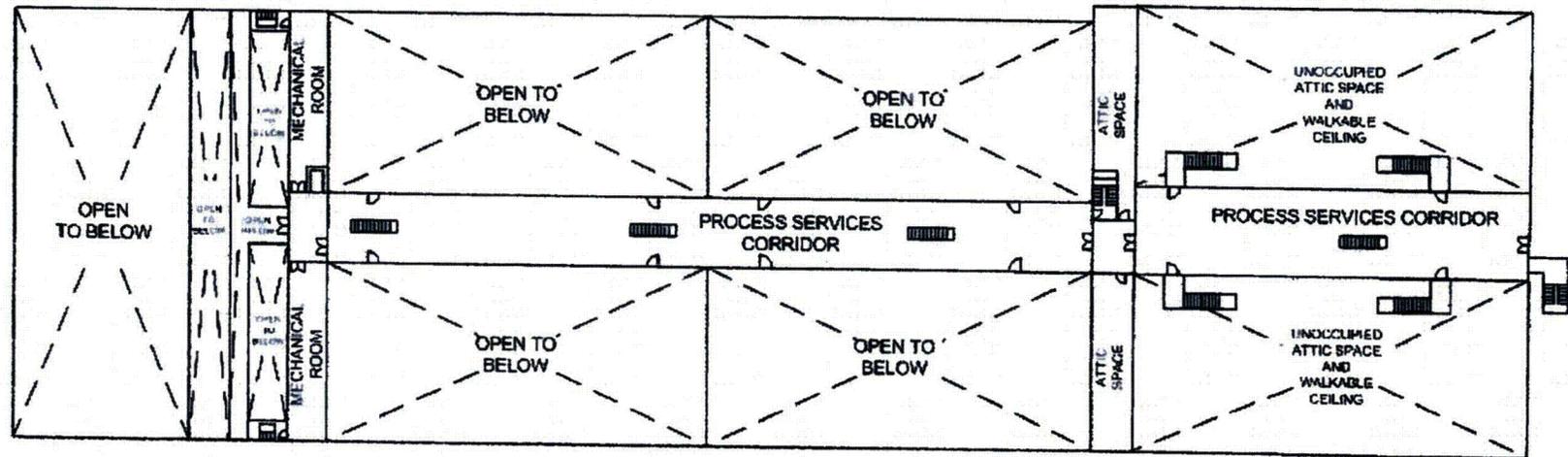
Figure 1.1-6 Separations Building Module Second Floor

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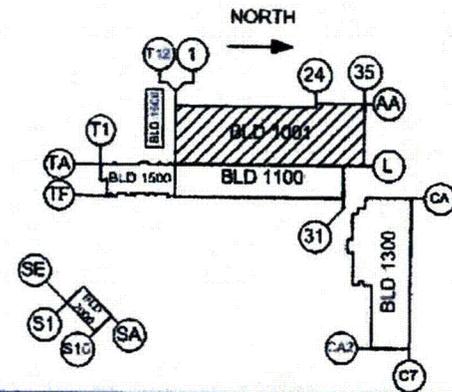
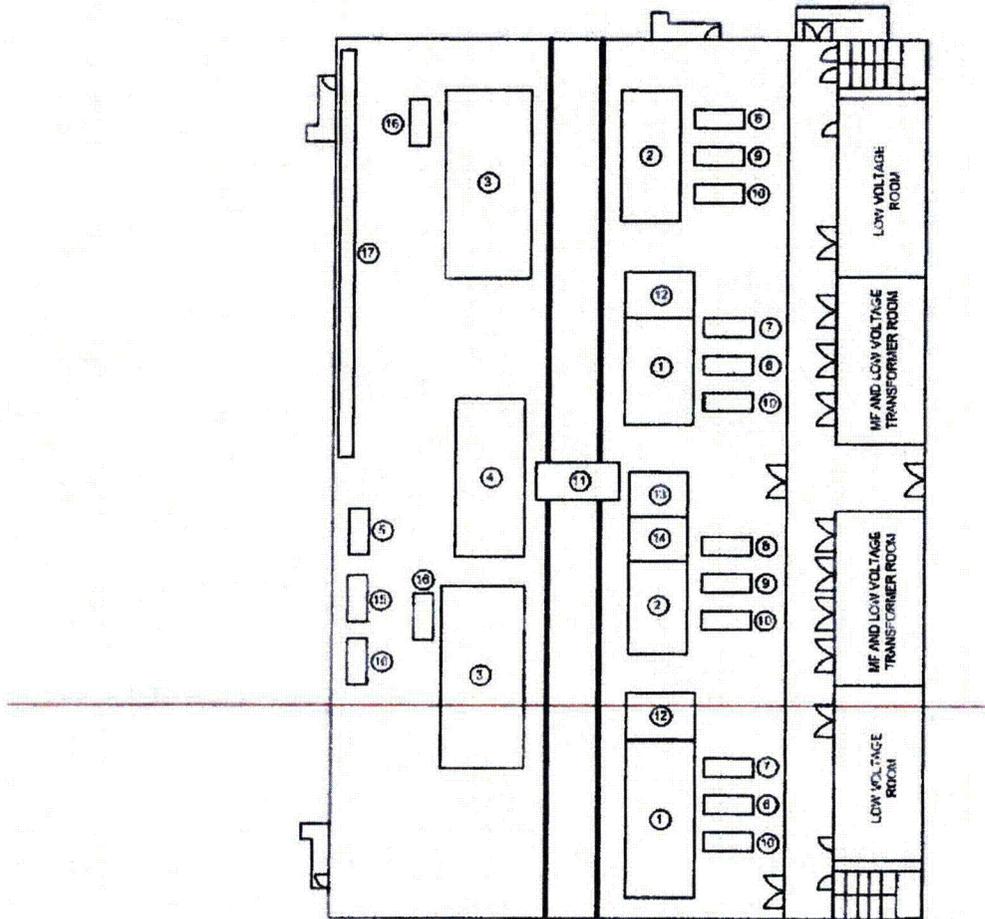
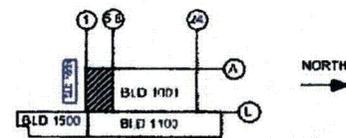


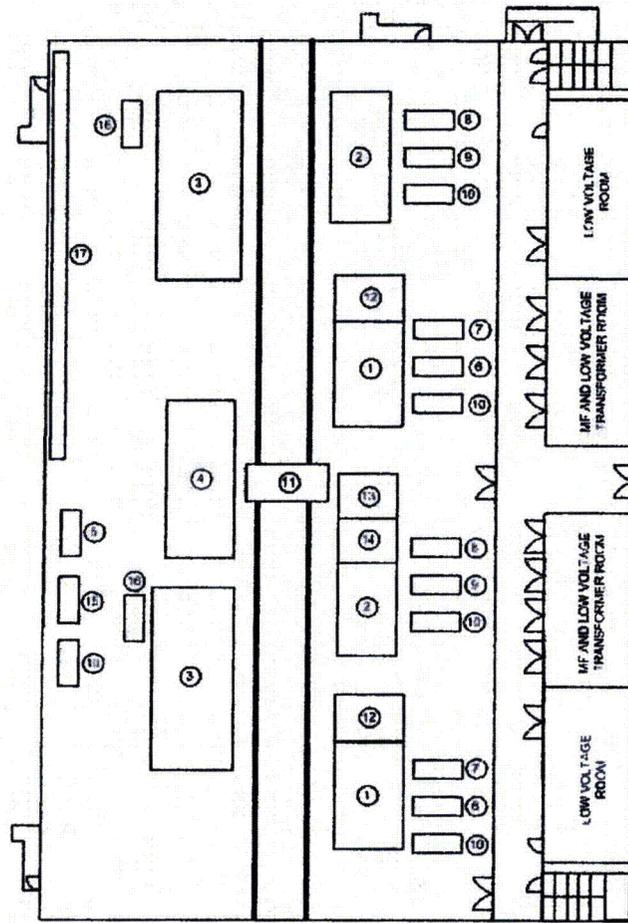
Figure 1.1-7 Separations Building Module Third Floor



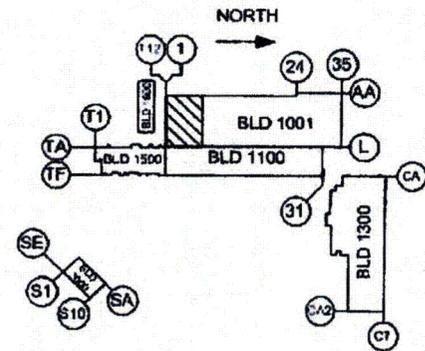
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- 1 SOLID FEED STATIONS
- 2 PRODUCT STATIONS
- 3 TAILS STATIONS
- 4 AUTOCLAVES
- 5 DONOR RECEIVER VACUUM PUMP TRAP SET
- 6 FEED PURIFICATION COLD TRAP
- 7 FEED PURIFICATION VACUUM PUMP TRAP SET
- 8 PRODUCT VACUUM PUMP TRAP SET
- 9 PRODUCT VENT COLD TRAP
- 10 COLD TRAP HEATER/CHILLER SET
- 11 RAIL TRANSPORTER
- 12 FEED PURIFICATION STATIONS
- 13 DONOR STATIONS
- 14 RECEIVER STATIONS
- 15 DONOR RECEIVER COLD TRAP
- 16 TAILS VACUUM PUMP TRAP SET
- 17 PUMPED EXTRACT GEVS





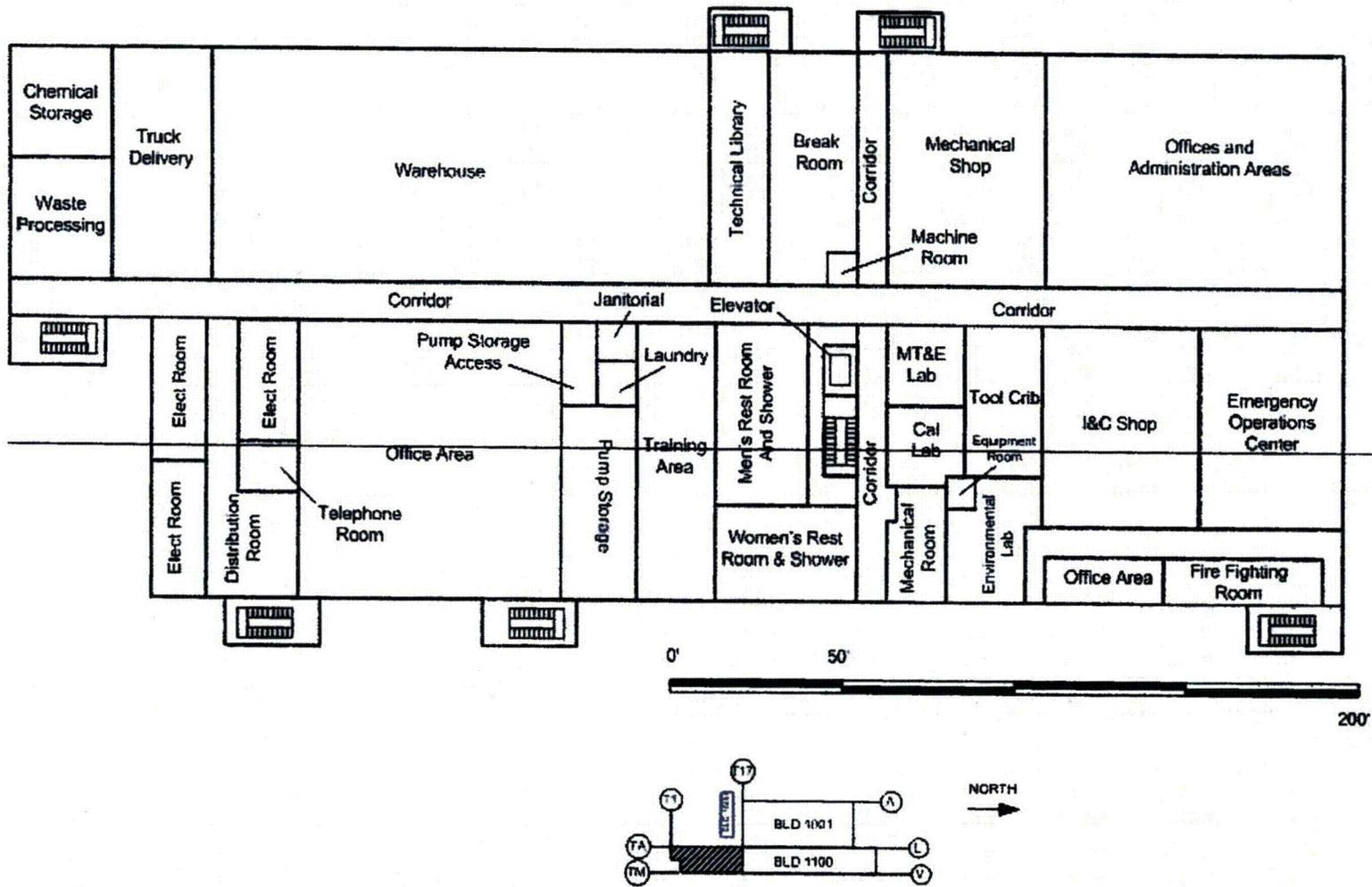
- 1 SOLID FEED STATIONS
- 2 PRODUCT STATIONS
- 3 TAILS STATIONS
- 4 AUTOCLAVES
- 5 DONOR RECEIVER VACUUM PUMP TRAP SET
- 6 FEED PURIFICATION COLD TRAP
- 7 FEED PURIFICATION VACUUM PUMP TRAP SET
- 8 PRODUCT VACUUM PUMP TRAP SET
- 9 PRODUCT VENT COLD TRAP
- 10 COLD TRAP HEATER/CHILLER SET
- 11 RAIL TRANSPORTER
- 12 FEED PURIFICATION STATIONS
- 13 DONOR STATIONS
- 14 RECEIVER STATIONS
- 15 DONOR RECEIVER COLD TRAP
- 16 TAILS VACUUM PUMP TRAP SET
- 17 PUMPED EXTRACT GEVS



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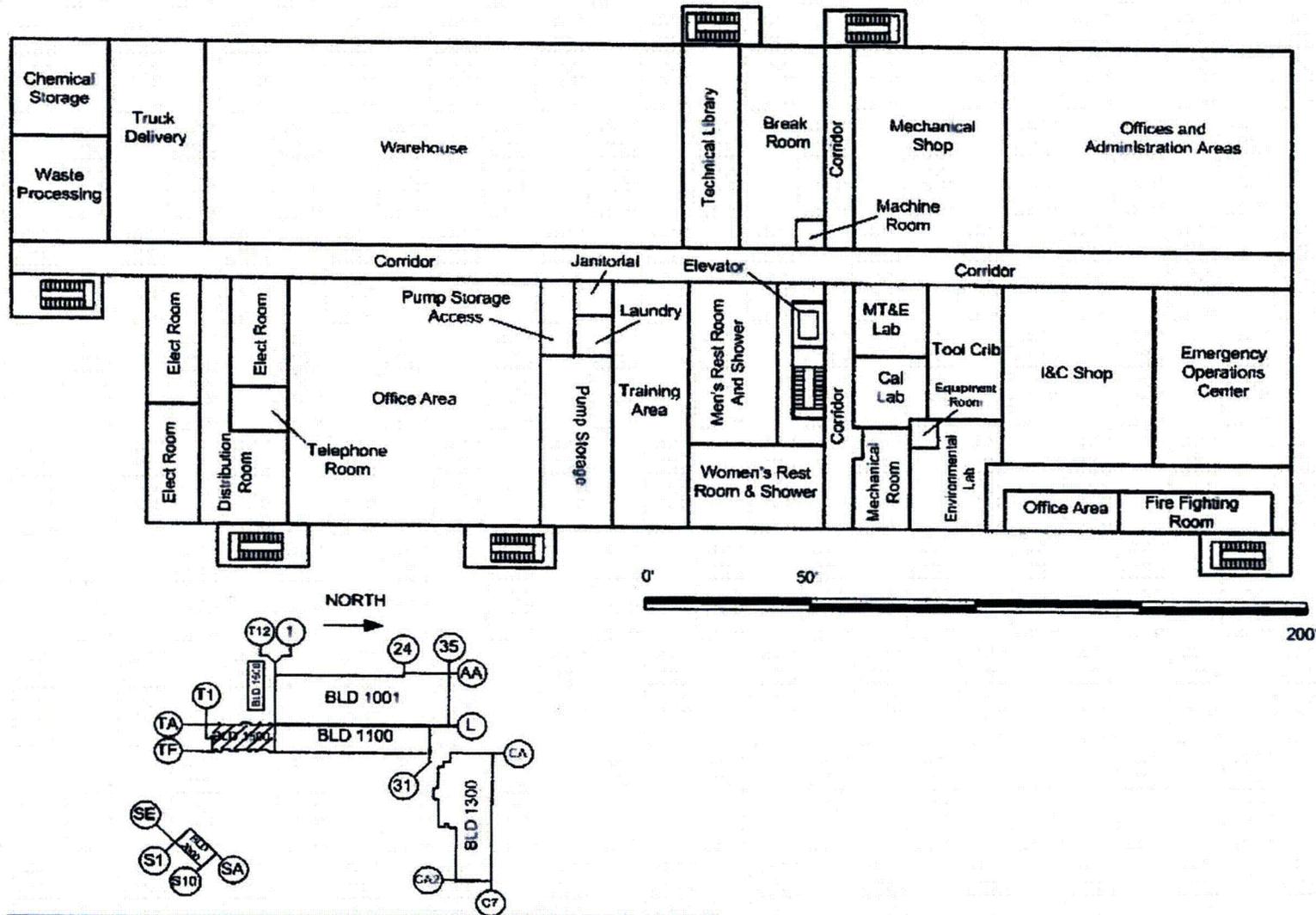
Figure 1.1-8 UF₆ Handling Area Equipment Location

1.6 Chapter 1 Figures



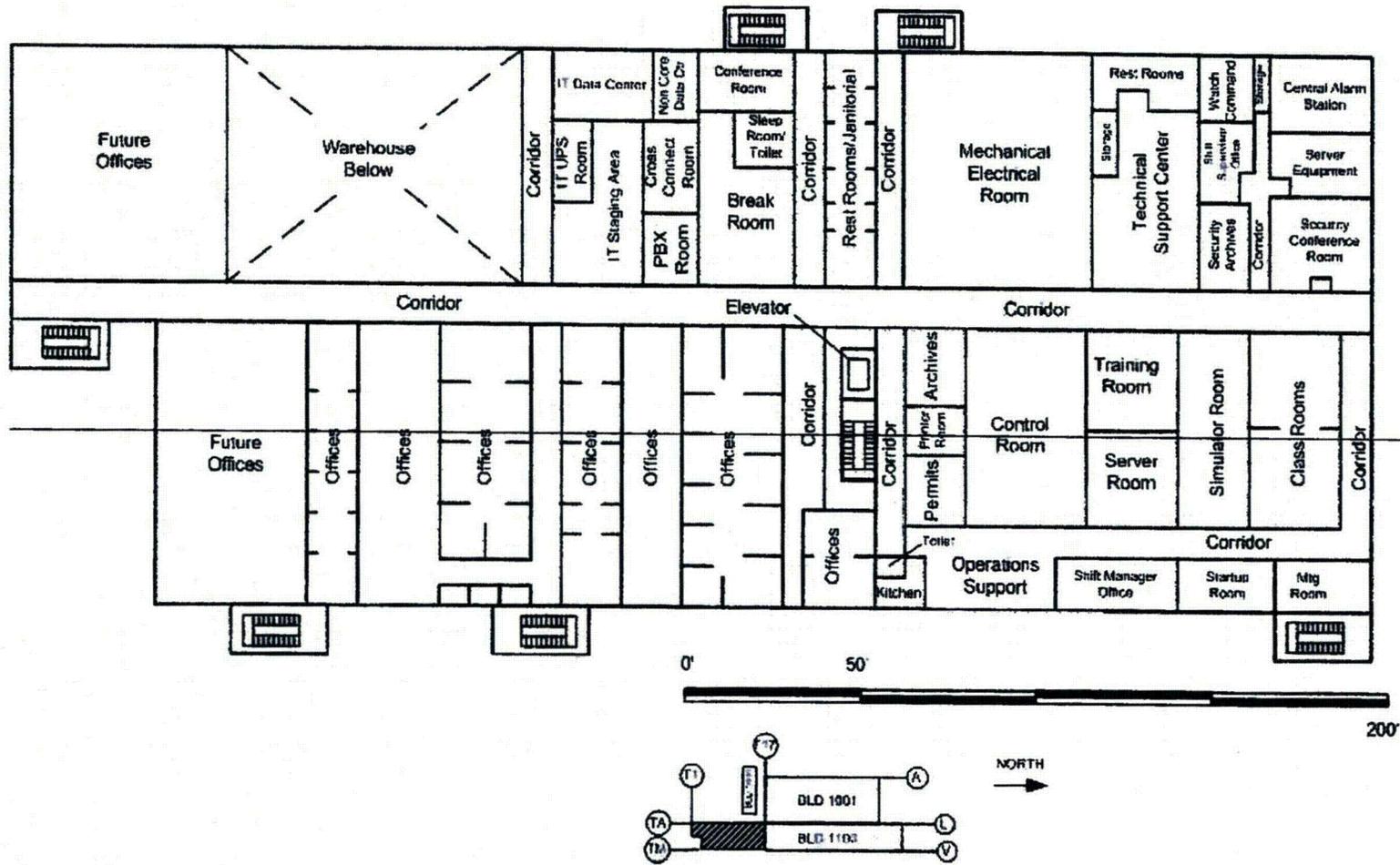
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1.6 Chapter 1 Figures

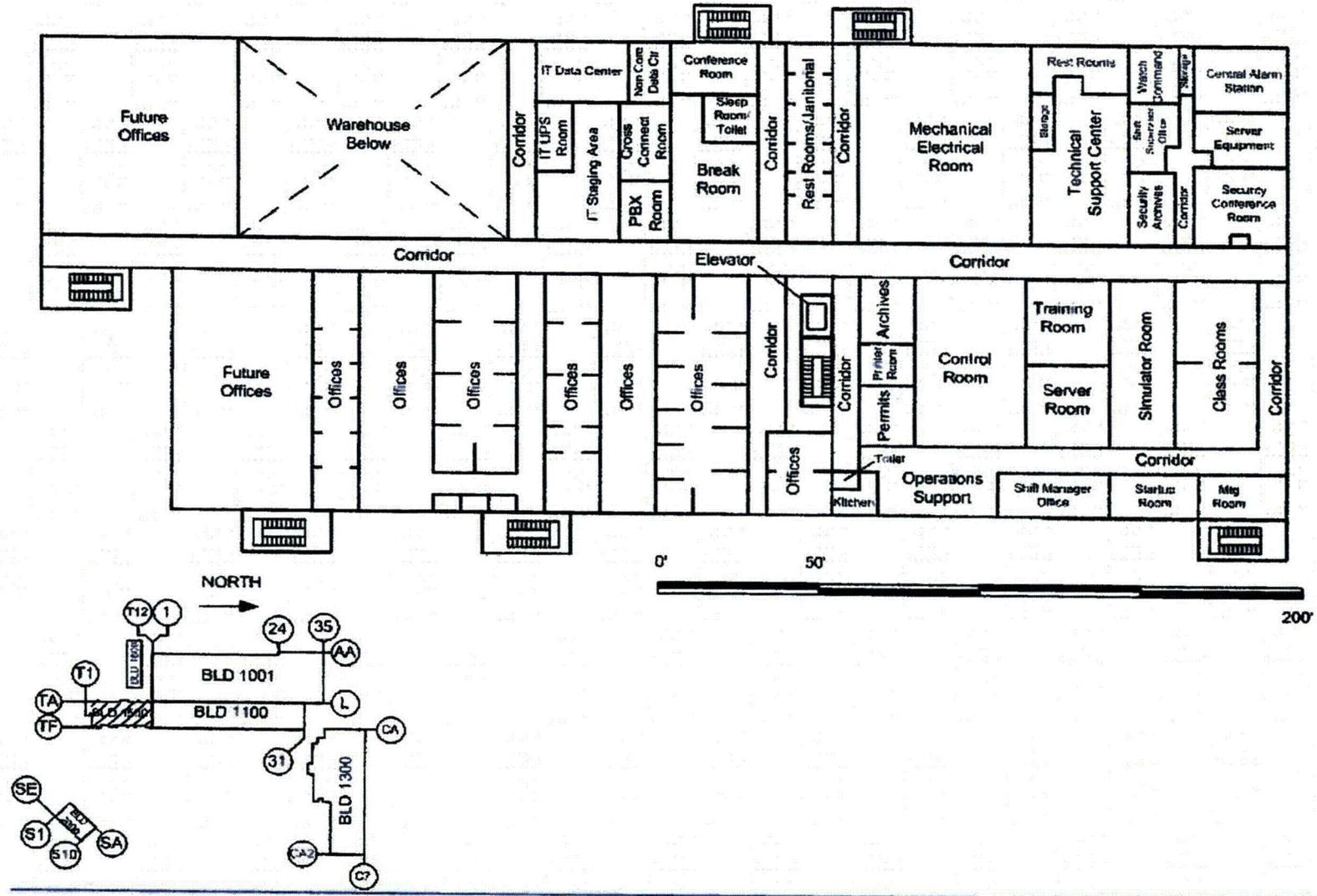


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Figure 1.1-9 Technical Services Building First Floor



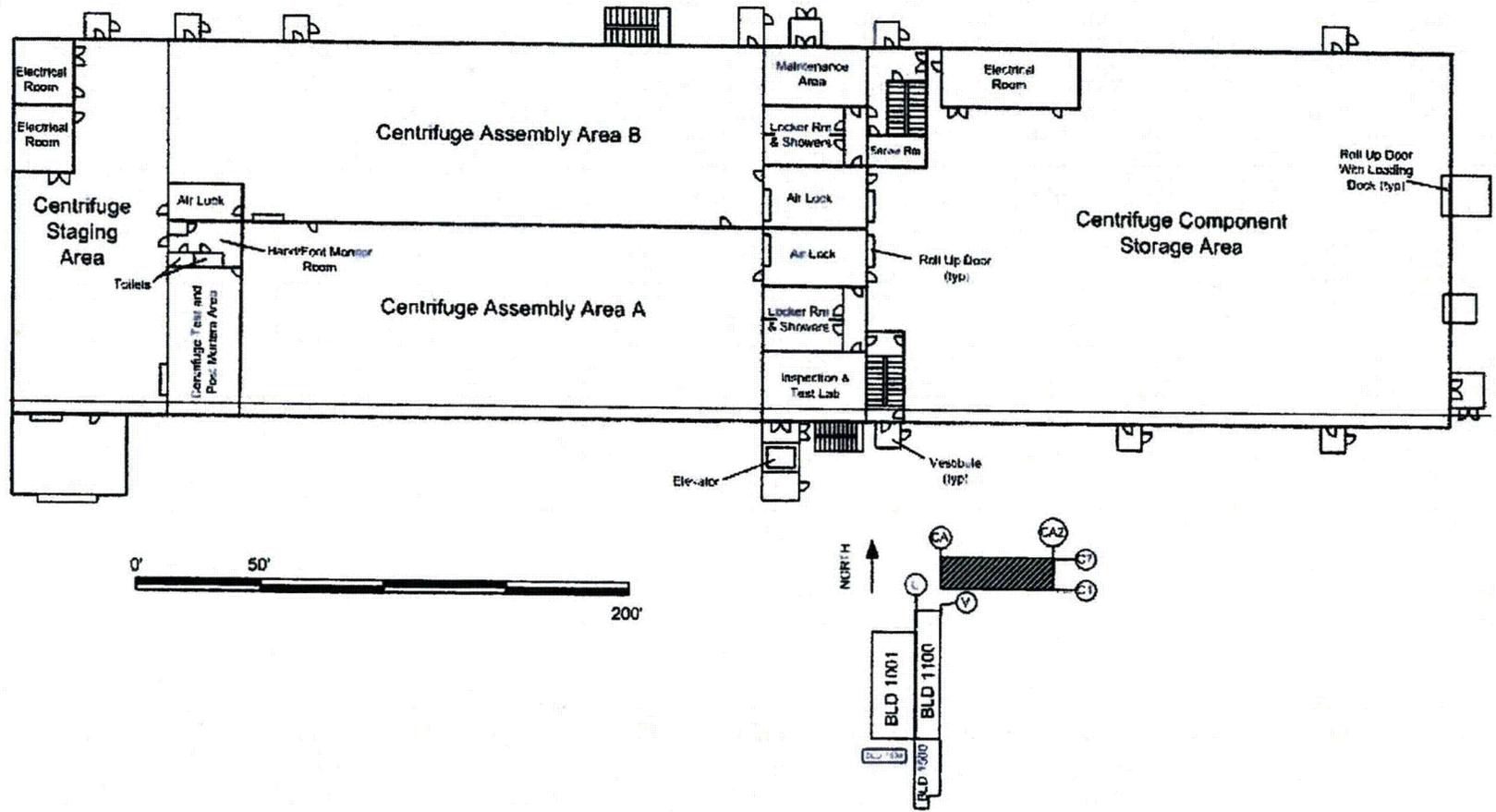
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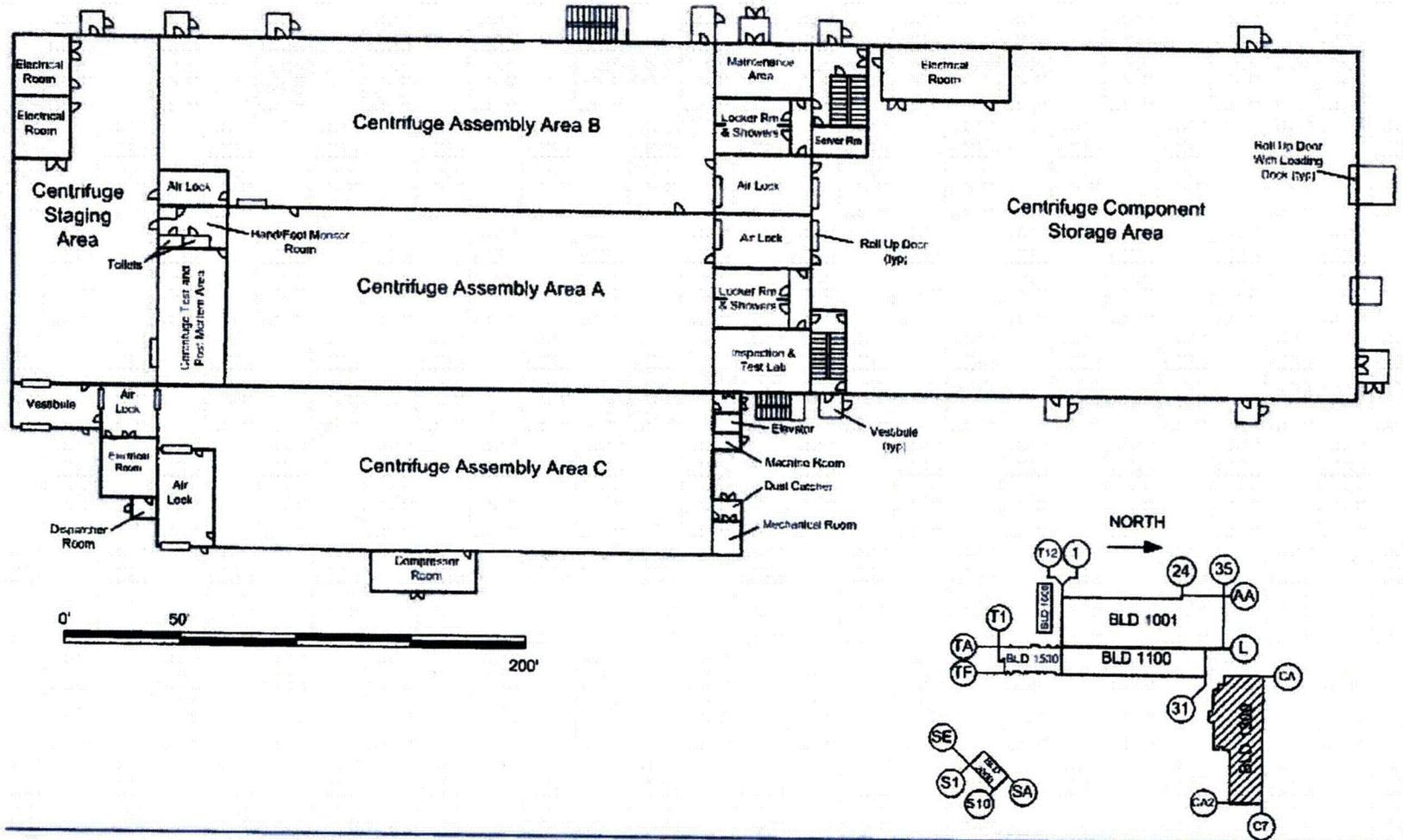
Figure 1.1-10 Technical Services Building Second Floor

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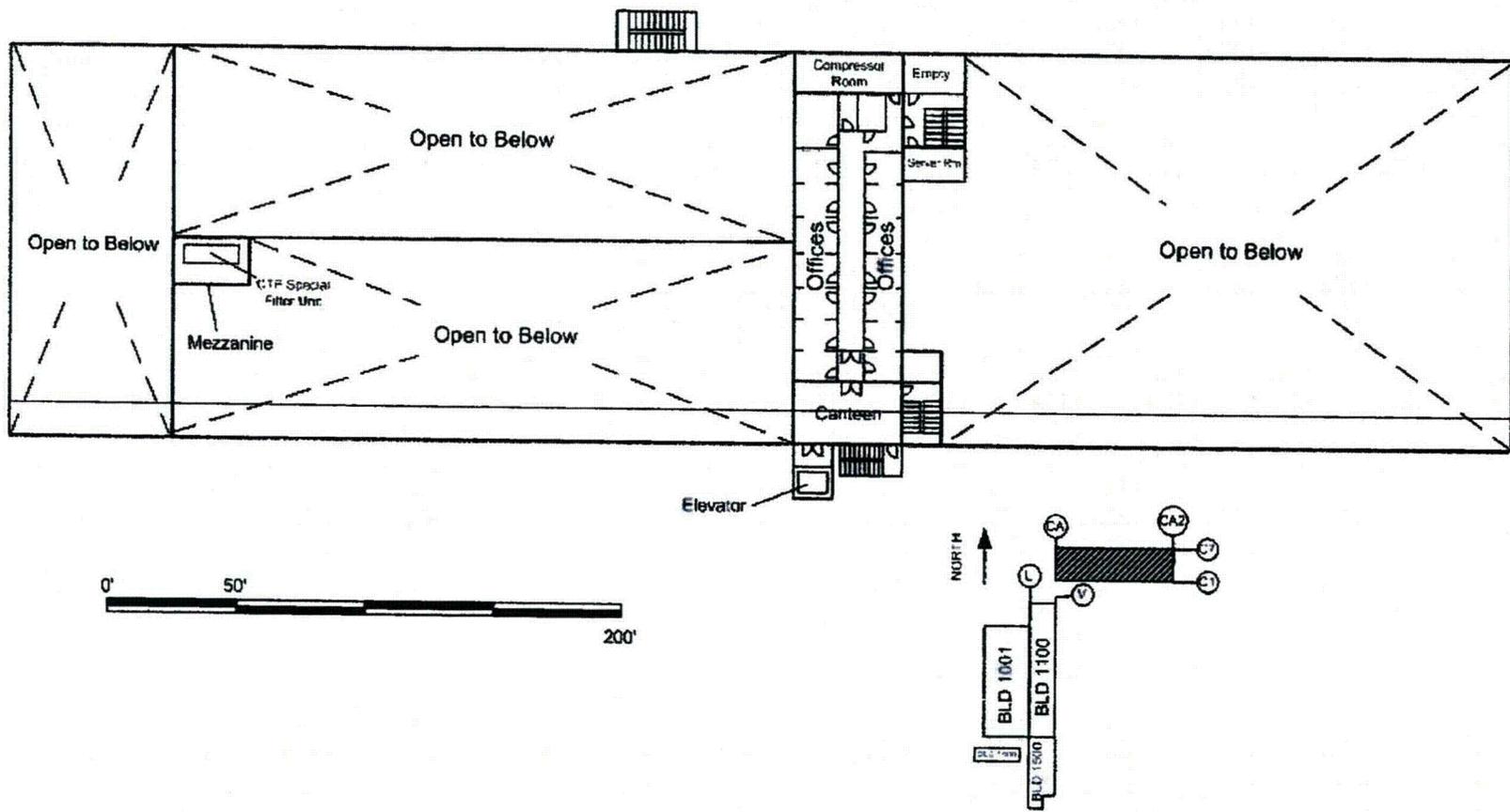
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1.6 Chapter 1 Figures

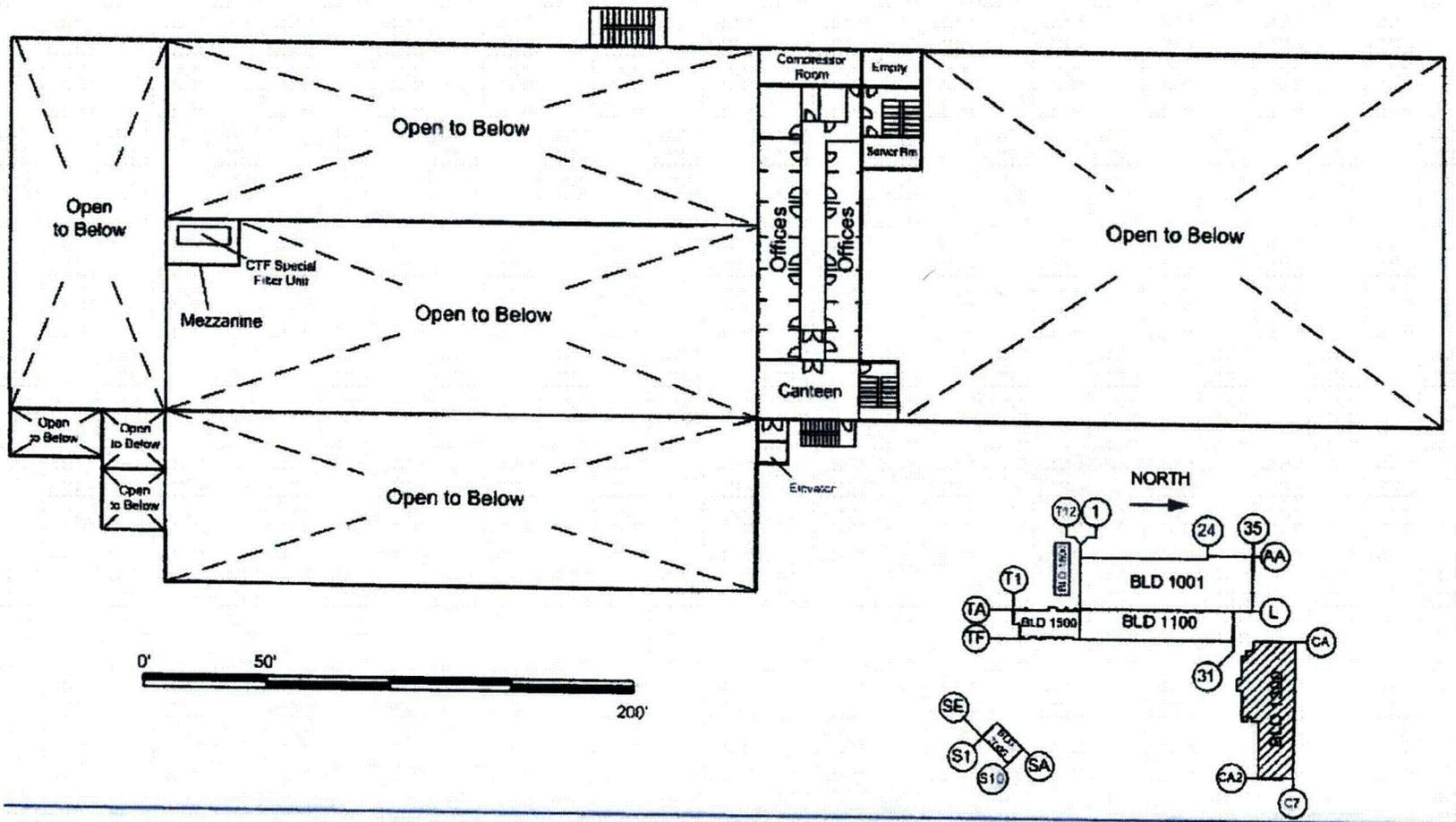


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Figure 1.1-11 Centrifuge Assembly Building First Floor



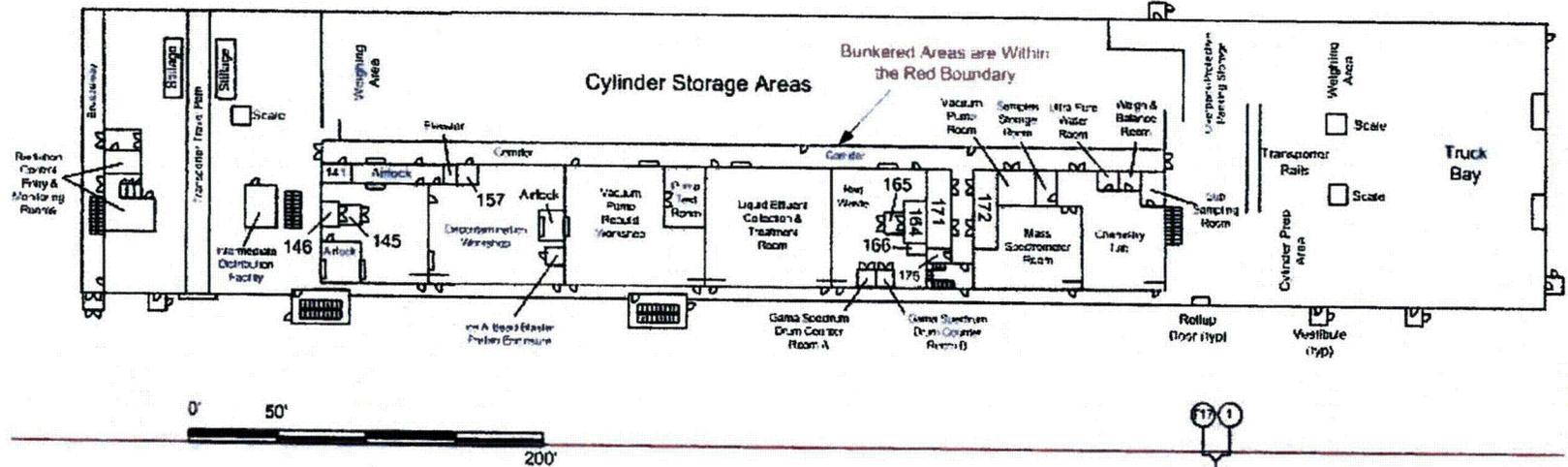
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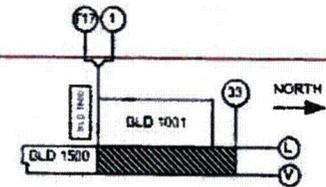
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Figure 1.1-12 Centrifuge Assembly Building Second Floor

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Room No.	Description
141	Personnel Decontamination Room
145	Cold Trap and Filter Unit Enclosure Trap Emptying and Drum Tipper Enclosure
146	Enclosure
157	Elevator Equipment Room
164	Drum Repackaging Enclosure
165	Whole Body Monitor Enclosure
166	Radioactive Source Storage Enclosure
171	Electrical Equipment Room A
172	Electrical Equipment Room B
175	Electrical and Mechanical Chase



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1.6 Chapter 1 Figures

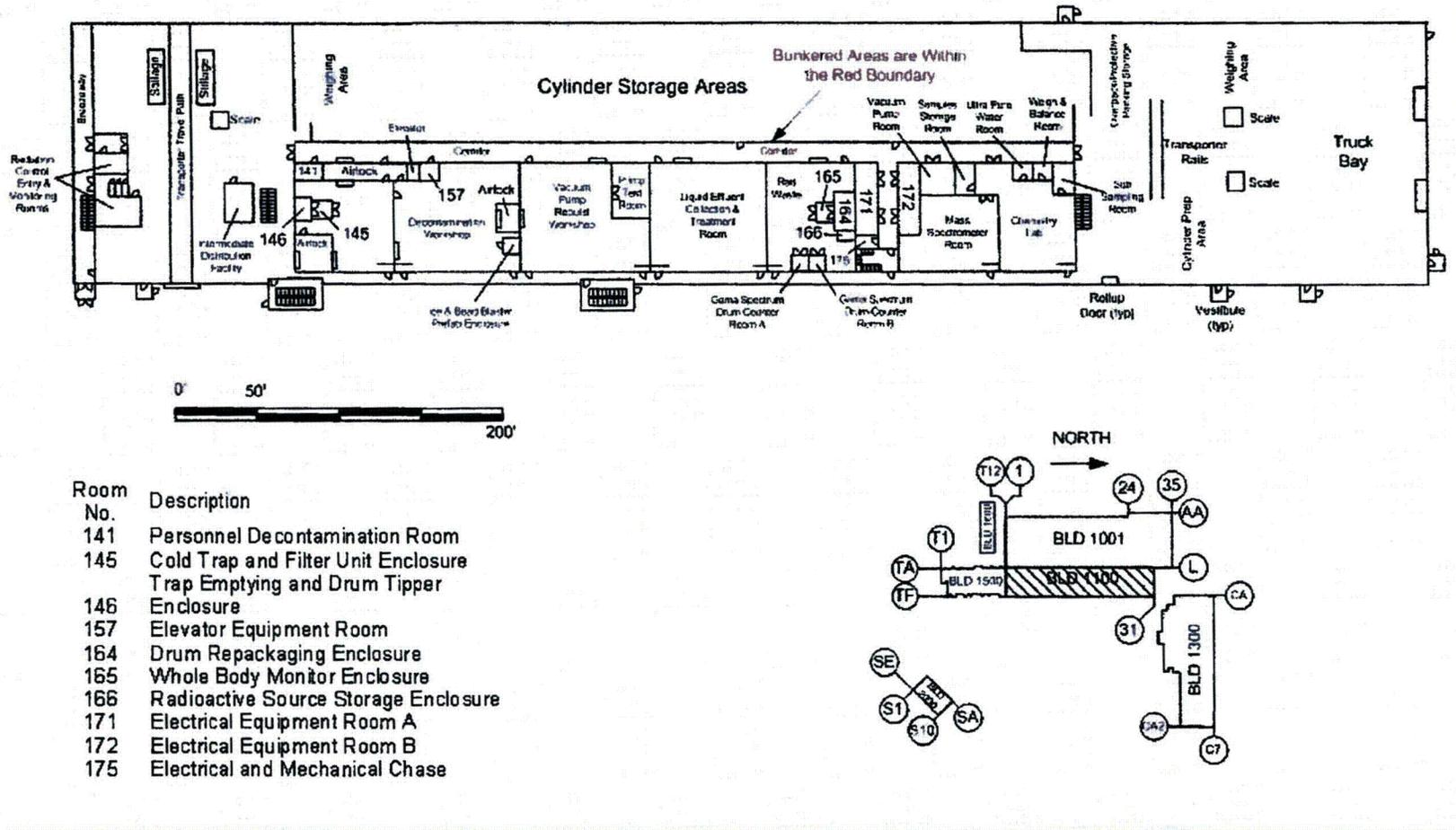
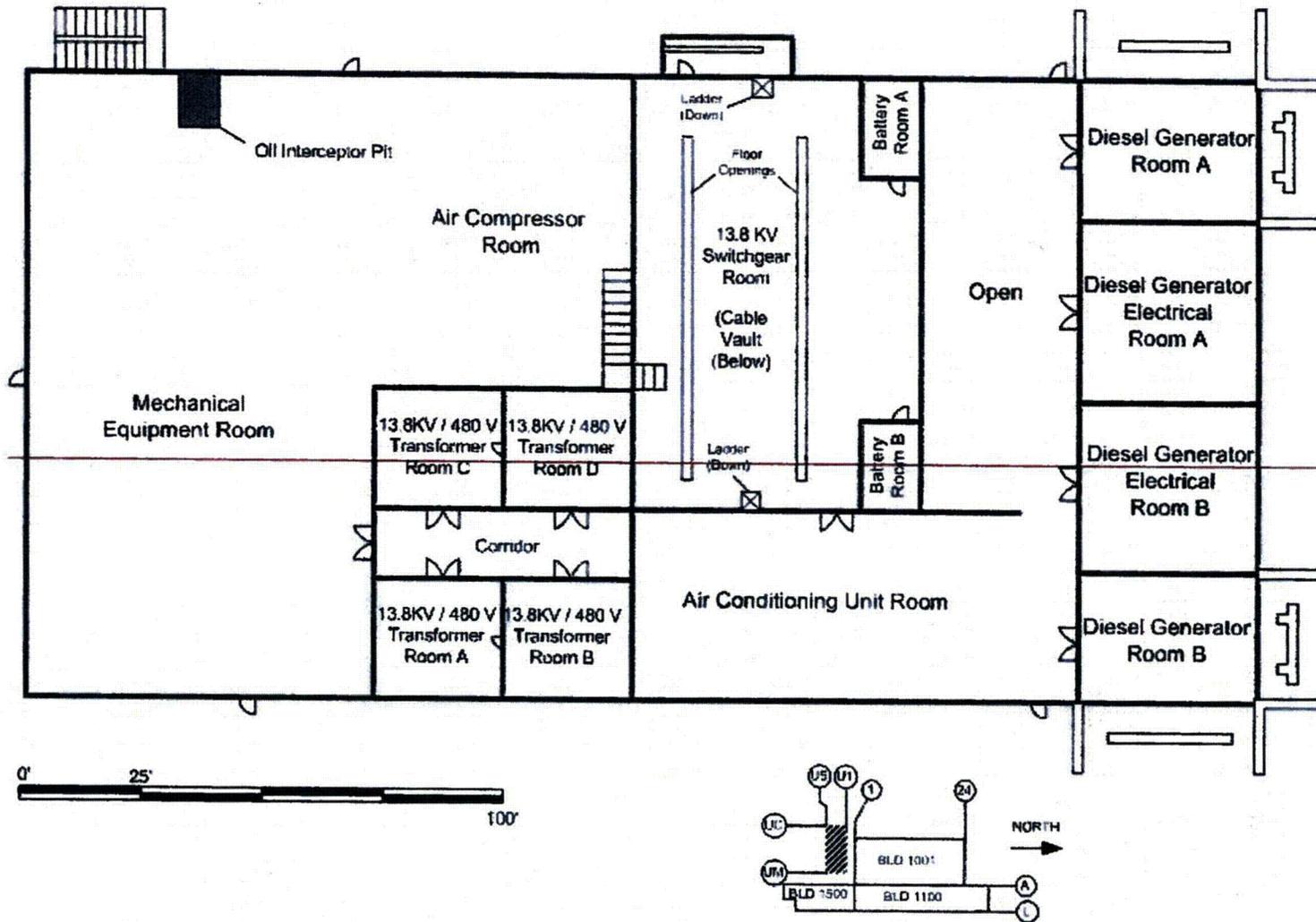
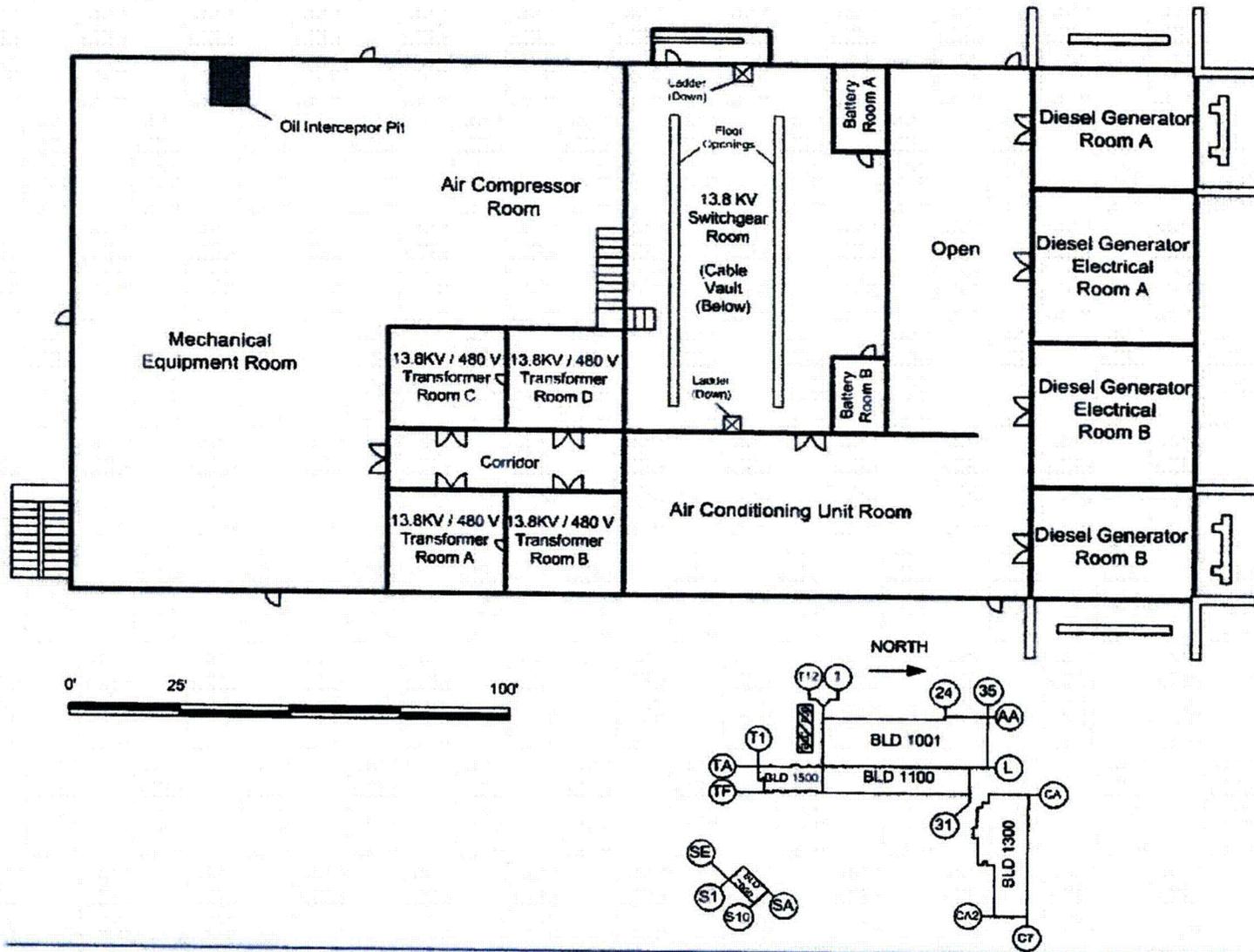


Figure 1.1-13 Cylinder Receipt and Dispatch Building First Floor



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Figure 1.1-14 Central Utilities Building First Floor

2.1 Organizational Structure

2.1 ORGANIZATIONAL STRUCTURE

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

2.1.1 Corporate Functions, Responsibilities, and Authorities

LES is a registered limited liability company formed solely to provide uranium enrichment services for commercial nuclear power plants. The LES company organization and management structure is described in Chapter 1, Section 1.2, Institutional Information.

LES has presented to Lea County, New Mexico a proposal to develop the NEF. Lea County would issue its Industrial Revenue Bond (National Enrichment Facility Project) Series 2004 in the maximum aggregate principal amount of \$1,800,000,000 to accomplish the acquisition, construction and installation of the project pursuant to the County Industrial Revenue Bond Act, Chapter 4, Article 59 NMSA 1978 Compilation, as amended. The Project is comprised of the land, buildings, and equipment.

Under the Act, Lea County is authorized to acquire industrial revenue projects to be located within Lea County but outside the boundaries of any incorporated municipality for the purpose of promoting industry and trade by inducing manufacturing, industrial and commercial enterprises to locate or expand in the State of New Mexico, and for promoting a sound and proper balance in the State of New Mexico between agriculture, commerce, and industry. Lea County will lease the project to LES, and LES will be responsible for the construction and operation of the facility. Upon expiration of the Bond after 30 years, LES will purchase the project.

The County has no power under the Act to operate the project as a business or otherwise or to use or acquire the project property for any purpose, except as lessor thereof under the terms of the lease.

In the exercise of any remedies provided in the lease, the County shall not take any action at law or in equity that could result in the Issuer obtaining possession of the project property or operating the project as a business or otherwise.

LES is responsible for the design, quality assurance, construction, operation, and decommissioning of the enrichment facility. The President of LES reports to the LES Board of Managers as described in Section 1.2.

The President receives policy direction from the LES Board of Managers. Reporting to the President is the Chief Operating Officer & Chief Nuclear Officer. The Vice President - ~~Engineering Project~~, Vice President - Operations, Vice President - ~~Construction and the Quality & Regulatory Affairs Director~~ Compliance and General Counsel all report to the Chief Operating Officer & Chief Nuclear Officer. The Quality Assurance Manager reports to the Quality & Regulatory Affairs Director for functional day to day activities and has a direct line of communication to the Chief Operating Officer & Chief Nuclear Officer for stop work authority. The Health & Safety Manager reports to the Plant Support Director which reports to the Vice President of Operations. The H&S Manager, Environmental Compliance Officer and Plant Support Director have a direct line of communication to the Chief Operating Officer & Chief Nuclear Officer for all matters concerning safety during operations, design and construction.

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2.1 Organizational Structure

Figure 2.1-1, LES Corporate, Design and Construction Organization shows the authority and lines of communication.

2.1.2 Design and Construction Organization

As the owner of the enrichment technology and operator of the enrichment facilities in Europe, LES has contracted Urenco Limited to prepare the reference design for the facility, while an architect/engineering (A/E) has been contracted to further specify structures and systems of the facility, and ensure the reference design meets all applicable U.S. codes and standards. A contractor specializing in site evaluations has been contracted to perform the site selection evaluation. A nuclear consulting company has been contracted to conduct the site characterization, perform the Integrated Safety Analysis and to support development of the license application.

During the construction phase, preparation of construction documents and construction itself are contracted to qualified contractors. The Vice President of ~~Construction Project~~ is responsible for managing, construction and construction turnover testing activities. The Vice President of Engineering has overall design responsibility, reports to the Vice President of Project and is the responsible design authority during construction. The Procurement Director is responsible for the procurement. Contractor QA Programs will be reviewed by LES QA and must be approved before work can start.

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Urenco will design, manufacture and deliver to the site the centrifuges necessary for facility operation. In addition, Urenco is supplying technical assistance and consultation for the facility. Urenco has extensive experience in the gas centrifuge uranium enrichment process since it operates three gas centrifuge uranium enrichment plants in Europe. Urenco is conducting technical reviews of the design activities to ensure the design of the enrichment facility is in accordance with the Urenco reference design information.

Procurement activities are coordinated by the LES Procurement Director. For procurement involving the use of vendors located outside the U.S., LES selects vendors only after a determination that their quality assurance programs meet the LES requirements. Any components supplied to LES are designed to meet applicable domestic industry code requirements or their equivalents as stated by the equipment specifications. The Procurement Director reports directly to the Chief Financial Officer and for quality and technical matters which reports to the Chief Operating Officer and Chief Nuclear Officer.

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The Vice President of ~~Construction Project~~ is responsible for managing the work and contracts. The lines of communication of key management positions within the engineering and construction organization are shown in Figure 2.1-1.

Position descriptions of key management personnel in the design and construction organization will be accessible to all affected personnel and the NRC.

2.1 Organizational Structure

2.1.3 Operating Organization

The operating organization for LES is shown in Figures 2.1-1, and 2.1-2, LES National Enrichment Facility Operating Organization. LES has direct responsibility for preoperational testing, initial start-up, operation and maintenance of the facility.

The Vice President – Operations is the Plant Manager, and reports to the Chief Operating Officer & Chief Nuclear Officer. The Plant Manager is responsible for the overall operation and administration of the enrichment facility after formal turnover from Construction and acceptance by Operations. He is also responsible for ensuring the facility complies with all applicable regulatory requirements. In the discharge of these responsibilities, the Plant Manager directs the activities of the following groups:

- Security
- Operations
- Technical Services
- Plant Support
- Commissioning & Acceptance

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

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

2.1.4 Transition From Design and Construction to Operations

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

The National Enrichment Facility (NEF) will commence operating when the first cascade (Cascade 101) is commissioned and placed into service. Construction activities will continue as each subsequent cascade is commissioned and placed into service. Due to the process system modular design, each cascade can be isolated from one another. This allows the construction, commissioning and operation of new cascades as well as the removal and replacement of existing centrifuges/cascades to continue while the remaining cascades are in operation. This modular design approach also supports the addition of subsequent Separations Building Modules (SBM) and extension modules with cascades in operation.

~~Towards the end of construction~~As the facility nears operation of the first cascade, the focus of the organization will shift from design and construction to construction turnover, initial start-up and operation of the each facility system. As the facility nears completion, LES will staff the LES NEF Operating Organization to ensure smooth transition from construction activities to operation activities after formal turnover of Design Authorities from the Vice President of Engineering. During this transition, the Health and Safety Manager, Environmental Compliance Officer and Plant Support Director have the authority to report safety concerns directly to the Chief Operating Officer & Chief Nuclear Officer (as shown in Figure 2.1-1 and Figure 2.1-2) for HS&E matters related to operations, design or construction. These positions are intentionally provided stop work authority at the Chief Operating Officer & Chief Nuclear Officer level to

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2.2 Key Management Positions

development and resolution of key technical issues, approving the NEF approved design, and establishing processes for design and configuration control. During the operations phase, after turnover, this also includes technical support for facility modifications (including administration of the configuration management system) and design support for operations and maintenance. Other responsibilities that reside solely with the Technical Services Director include facility management (facility maintenance, warehouse management, and outsourced maintenance supervision), and contamination control (decontamination and waste treatment). The Technical Services Director is also responsible for records management. In the event of the absence of the Plant Manager, the Technical Services Director may assume the responsibilities and authorities of the Plant Manager.

G. Plant Support Director

The Plant Support Director reports to the Plant Manager and has the responsibility for emergency planning; ensuring training is provided for facility employees as well as implementation of the Criticality Safety Program. In addition, the Plant Support Director maintains a line of communications with the Radiation Production Manager. In doing so he is ensuring proper contamination control and nuclear criticality safety protection. The Plant Support Director is also responsible for the fire protection program, industrial safety, chemical safety and material accountability program. The Plant Support Director, in coordination with the Community Affairs Director, has the responsibility for providing information about the facility and LES to the public and media, including ensuring that the public and media receive accurate and up-to-date information during an abnormal event at the facility. In the event of the absence of the Plant Manager, the Plant Support Director may assume the responsibilities and authorities of the Plant Manager.

This position has a line of communications to the Chief Operating Officer and Chief Nuclear Officer to ensure objective nuclear safety audit, review, and control activities are maintained. This position is intentionally provided stop work authority at the Chief Operating Officer & Chief Nuclear Officer level to provide significant continued focus on the health, safety, and environment goals during design and construction when the operating organization is not yet fully developed and implemented.

H. Commissioning & Acceptance Director

The Commissioning & Acceptance Director reports to the Plant Manager and has the responsibility for the implementation of major facility modifications and acceptance of the facility during commissioning. In the event of the absence of the Plant Manager, the Commissioning & ~~Plant Control~~ Acceptance Director may assume the responsibilities and authorities of the Plant Manager.

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I. Performance Assessment and Feedback Manager

The Performance Assessment and Feedback Manager reports to the Quality & Regulatory Affairs Director and has the responsibility for organizational performance metrics, and implementing the Corrective Action Program (CAP), Nonconformance Process and Industry Experience Program.

2.2 Key Management Positions

During emergency conditions the 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.

In matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager.

Q. Industrial Safety Officer

The Industrial Safety Officer reports to the Health and Safety Manager and has the responsibility for the implementation of facility industrial safety programs and procedures. This shall include programs and procedures for training individuals in safety. The Industrial Safety Officer is also responsible for the preparation and/or review of chemical safety programs and procedures for the facility.

R. Fire Protection Officer

The Fire Protection Officer reports to the Plant Support Director and has the responsibility for maintaining the performance of the facility fire protection systems.

S. Criticality Safety Officer

Criticality Safety Officer reports to the Health and Safety Manager and is responsible for implementing the Criticality Safety Program in the operating organization, including ~~conducting and reporting~~ ensuring that periodic nuclear criticality safety assessments are performed and reported.

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T. Criticality Safety Engineers

Criticality Safety Engineers report to the Plant Support Director and are responsible for the preparation and/or review of nuclear safety criticality evaluations and analysis. Nuclear criticality safety evaluations and analyses require independent review by a second Criticality Safety Engineer.

U. Deleted

V. Shift Operations Manager

The Shift Operations Manager reports to the Operations Director, and has the responsibility of directing the day-to-day operation of the facility. This includes such activities as ensuring the correct and safe operation of UF₆ processes, proper handling of UF₆, and the identification and mitigation of any off normal operating conditions.

Cascade Criticality Control

The design for the URENCO USA facility is based on the design and operating experience of our URENCO enrichment facilities in Europe, which includes more than 30 years of experience and numerous assay units. Experts from our European URENCO and ETC organizations were fully engaged in our review of accident sequences and analysis to ensure that we accurately addressed the risks associated with potential criticality hazards.

The key factor affecting the enrichment level is the ratio of feed to product flow. The higher the ratio is, the higher the enrichment is. A high ratio can be achieved by one of the following:

- Increase in feed flow
- Decrease in product flow
- A combination of the above two

The enrichment controls for the URENCO USA cascades are the control devices for the feed and product systems. The control system is designed to provide a precise enrichment output to ensure that the desired enrichment is produced for our customers. The control system provides alarms to alert the operator if the required pressures or control device positions deviate from the operating band set for that campaign. In addition to the alarms monitoring the process parameters, the Plant Control System (PCS) will also alarm with the failure of a cascade PLC or the plant control system. Routine sampling and monitoring is performed to ensure that the product output meets the expected enrichment for the campaign. In addition, specific sampling is performed to ensure that the correct enrichment is obtained after changing the settings for a new campaign or planned change of enrichment. The PCS identifies and advises of adverse conditions and protects plant assets. The system is not credited for performing actions important to safety. The ISA Summary, section 3.5.9.3 states:

“The PCS is required for remote operation, production protection and asset protection. Protection is carried out by local control and protection systems of the individual process equipment. The PCS is not required and not relied upon for protection of the public or environment; therefore, the PCS is not an IROFS.”

The NEF process is licensed to operate at assays below 5wt% ²³⁵U. The plant has been analyzed and shown to be safe operating at assays of 6wt% ²³⁵U. This is based on $K_{eff} < 0.95$ (where $K_{eff} = k_{calc} + 3\sigma_{calc}$). The design of product control devices incorporates a passive engineered mechanical stop to prevent excessive enrichments being produced. The mechanical stop is not adjustable and requires a design modification to change. To allow operational flexibility and satisfy the design function, this mechanical stop design could allow enrichments slightly above 6wt% ²³⁵U as it accounts for inaccuracies in settings and optimal performance ranges for the cascades. It is important to note that the 5% operational and 6% analytical limits are established to support safe operation within established license conditions. However, exceeding 10 CFR 70.61 for the purposes of needing IROFS to prevent criticality is based on $K_{eff} < 0.95$ as described above. If the correct combination of multiple failures could occur such that 5%, or

that the software will support operating in our operating license limits for the initial operation of the software. In addition we verify that the first cascade that receives the new software is monitored and tested to ensure that the cascade is producing the correct enrichment prior to loading the software onto the other cascades. This provides a high level of confidence that the software change will not be a common mode challenge. Monitoring and sampling on the original cascade product enrichment ensures that the output of the cascade is correct prior to loading the software into subsequent cascades. Monitoring and sampling of subsequent cascade and assay enrichment also ensures that there are no common mode issues that can result in over-enrichment. These measures make the risk for a criticality a non-credible event. This is independent of the PCS or the PLC in the cascade system.

Addressing the second cause for item 2 above, it is feasible that during preparation for a new campaign, the new settings for the cascades could be incorrectly calculated or inputted to the system and result in causing the cascades to produce enrichments that are higher than intended. In the worst case, this could result in the valve stop being the limiting factor for the enrichment. As described above this would cause numerous alarms to occur. Even though there would be a substantial amount of time required in this condition to generate sufficient mass to provide a potential criticality and would have been checked numerous times with our sampling activities, IROFSC6 was developed to ensure that the correct settings were calculated and inputted for the desired cascade settings.

IROFSC6 is a sole enhanced administrative control (EAC), requiring independent verification for both calculation and setting of the cascade enrichment controls. Implementation of IROFSC6 consists of two parts:

- Calculate and independently verify the setting parameters including feed flow, product flow and cascade header pressure with CASCAL (QL-1 software) for input into the Plant Control System (PCS)
- Set and independently verify the settings for the cascade enrichment control devices – The principal devices are the feed control valve and product control valve on the cascade valve frame. Both valves are mechanical valves monitored by the Programmable Logic Controller (PLC) in the PCS, which controls the enrichment.

IROFSC6 not only ensures that the cascade is set for less than our license limit of 5%, it also ensures that we have our appropriate settings to support our next campaign. Following a change of enrichment, several assay samples are taken to confirm the change has produced the required assay. Since the mechanical stop limits enrichment to less than 6.35wt% 235U, the time taken before criticality safe parameters ($K_{\text{eff}} = 0.95$) are approached, due to coincident moderator ingress, is orders of magnitude longer than the time it takes to fill a product cylinder, even at the lowest fill rate (ETC4097220). It is not credible that the plant will operate with leak rates orders of magnitude greater than those used in the assessment of ETC4097220. Each individual product cylinder will therefore remain sub-critical. Also, product liquid sampling for each cylinder will indicate the assay enrichment as well as the monitoring and sampling of the assay. Comparison of these samples should each highlight any errors. Because the system runs at a extremely low vacuum, the quantity of U235 present, regardless of enrichment level, at any point in the system

Our ISA process was utilized and an analysis of the hazards associated with cascade criticality controls was performed. The team consisted of experts from our European URENCO and ETC organizations as well as from URENCO USA. The results of the analysis identified one potential credible scenario for criticality. IROFSC6 was developed to reduce the risk for this scenario. We have established a very conservative basis for the prevention of criticality for our Cascade system and have met the criteria as stated in 10 CFR 70.61, our Material License, and our licensing documents.

Alternative for Cascade Criticality Controls

Our position is that the only credible failure for control of criticality for the Cascade System is tied to human error with the calculation and input of the data for the new campaign. IROFS C6 is designed to address this issue. Our analysis supports the response of the other hazards in the HAZOP as non-credible.

As an alternative, we recommend that we re-instate IROFS C7 which institutes sampling and monitoring of enrichment to ensure that enrichment is correct and to identify if there are any unexpected changes in the enrichment output. The periodicity of the sampling or monitoring will be based on analysis to ensure that an unexpected change will be identified and actions are taken to prevent the potential of a critical condition occurring. The expected periodicity for sampling based on analysis from the ETC document, ETC4097220, "Risk of Criticality due to Over-Enrichment," is once or twice per week. A sample would also be taken following a change to the enrichment settings due to a new campaign and to the 1st cascade after any software changes to PCS. In addition to IROFS C6 which ensures that the correct settings are input to the cascade system for a new cascade, IROFS C7 would provide additional verification that the output enrichment for the cascade and assay are within expected operating bands and within our operating license limits. This will provide an independent and diverse verification that enrichments are correct.

This will also provide a means to ensure that we identify any enrichment issues independent of the PCS or the cascade PLC.

2.2 Key Management Positions

coordinating and maintaining testing programs for the facility, including the testing of systems and components to ensure the systems and components are functioning as specified in design documents.

CC. Security Manager

The Security Manager reports to the Vice President of Operations and has the responsibility for directing the activities of security personnel to ensure the physical protection of the facility. The Security Manager is also responsible for the protection of classified matter at the facility and obtaining security clearances for facility personnel and support personnel.

DD. Information Services Manager

The Information Services Manager reports to the ~~Facilities Manager~~ Technical Services Director and has the responsibility for adequately controlling documents at the facility.

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EE. Training Manager

The Training Manager reports to the Plant Support Director and has the responsibility for conducting training and maintaining training records for personnel at the facility.

FF. Procurement Director

The Procurement Director reports to the Chief Financial Officer and has the responsibility for ensuring spare parts and other materials needed for operation of the facility are ordered, received, inspected and stored properly. ~~For quality and technical matters the Procurement Director reports to the Chief Operating Officer & Chief Nuclear Officer.~~

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GG. Deputy Director of Operations

The Deputy Director of Operations reports to the Director of Operations and assists the Director of Operations and has the responsibility for Shift Operations, Operations Support, Logistics Services and Chemistry Services. This includes such activities as ensuring the correct and safe operation of UF₆ processes, proper handling of UF₆, and the identification and mitigation of any off normal operating conditions, UF₆ cylinder management (including transportation licensing), directing the scheduling of enrichment operations to ensure smooth production, ensuring proper material and equipment are available for the facility, developing and maintaining production schedules and procedures for enrichment services, ensuring that cylinders of uranium hexafluoride are received and routed correctly at the facility, all transportation licensing and plant and environmental analysis.

HH. Quality & Regulatory Affairs Director

The Quality and Regulatory Affairs Director reports to the Vice President Compliance/General Counsel and has responsibility for the direction of Quality Assurance, Performance Assessment and Feedback (including the Corrective Action Program) and Licensing activities (including the Industry Experience Program). The Quality & Regulatory Affairs Director has overall responsibility for the development of the LES QA Program. The Quality and Regulatory Affairs Director has responsibility for coordinating facility activities to evaluate and assist the LES

2.2 Key Management Positions

The Safety Review Committee shall meet at least once per calendar quarter.

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 Plant Manager and appropriate Managers within 30 days and be retained in accordance with the records management system.

2.2.4 Personnel Qualification Requirements

The minimum qualification requirements for the facility functions that are directly responsible for its safe operation shall be as outlined below consistent with NUREG-1520. This includes the facility manager (Plant Manager), Operations Manager, Shift Managers, and managers for various safety and environmental disciplines. The nuclear experience of each individual shall be determined to be acceptable by the Chief Operating Officer and Chief Nuclear Officer. "Responsible nuclear experience" for these positions shall include (a) responsibility for and contributions towards support of facility(s) in the nuclear fuel cycle (e.g., ~~design, construction, operation, and/or decommissioning~~ mining, milling, processing, conversion, enrichment, fuel fabrication, reactor use, storage, fuel processing or final disposition of waste), and (b) experience with chemical materials and/or processes. Relevant work experience of at least five years, in addition to the minimum experience requirements specified in the section, may be substituted for educational Bachelor's degree requirements. The Chief Operating Officer and Chief Nuclear Officer may approve different experience requirements for key positions. Approval of different requirements shall be done in writing and only on a case-by-case basis.

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The assignment of individuals to the Manager positions reporting directly to the Plant Manager, and to positions on the SRC, shall be approved by the Plant Manager. Assignments to all other staff positions shall be made within the normal administrative practices of the facility.

The actual qualifications of the individuals assigned to the key facility positions described in Section 2.2.1, Operating Organization will be maintained in the employee personnel files or other appropriate file at the facility. Development and maintenance of qualification records and training programs are the responsibility of the Training Manager.

2.2 Key Management Positions

G. Technical Services Director

The Technical Services Director 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. Plant Support Director

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

I. Emergency Preparedness Manager

The Emergency Preparedness Manager shall have a bachelor's degree (or equivalent) and a minimum of six years of experience in the implementation and supervision of emergency plans and procedures, at least three of which must be at a nuclear facility. No credit for academic training may be taken toward fulfilling this experience requirement.

J. Deleted

K. Environmental Compliance Officer

The Environmental Compliance Officer shall have a bachelor's degree (or equivalent) and a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear environmental compliance program.

L. Radiation Protection Manager

The Radiation Protection Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and three years of responsible nuclear experience associated with implementation of a Radiation Protection program.

M. Industrial Safety Officer

The Industrial Safety Officer shall have a minimum of two years experience in the preparation and/or review of chemical safety programs and procedures and shall have, as a minimum, a bachelor's degree (or equivalent) in either an engineering or a scientific field and three years of appropriate, responsible nuclear experience associated with implementation of a facility industrial and chemical safety program.

N. Criticality Safety Officer

Criticality Safety Officer (CSO) shall have experience in the implementation of a criticality safety program. This individual shall hold a Bachelor of Science or Bachelor of Arts degree in an engineering or scientific field and have successfully completed a training program, applicable to the scope of operations, in the physics of criticality and in associated safety practices. In addition, the CSO shall have at least two years of experience performing criticality safety analyses.

The CSO is a technical position with responsibility for oversight of the program. For this reason, the CSO shall have educational and experience requirements equal to or greater than those of a Criticality Safety Engineer as defined in Section 2.2.4.ON.

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2.3 Administration

- Functional testing.

These maintenance categories are discussed in detail in Chapter 11, Management Measures.

2.3.3 Training and Qualifications

Prescribed training programs shall be established for NEF employees. General Employee Training shall be provided to employees prior to receiving unescorted access, and shall address safety preparedness for all safety disciplines (criticality, radiological, chemical, industrial), ALARA practices, and emergency procedures. In-depth training programs shall be provided to individuals depending on job requirements in the areas of radiological safety (for all personnel with access to the ~~Restricted Area~~ Radiologically Controlled Area (RCA)) and in criticality safety control. Nuclear criticality safety training shall satisfy the recommendations of ANSI/ANS-8.20, Nuclear Criticality Safety Training. Continuing training of personnel previously trained shall be performed for radiological and criticality safety at least annually, and shall include updating and changes in required skills. The training program shall include methods for verifying training effectiveness, such as written tests, actual demonstration of skills, and where required by regulation, maintaining a current and valid license demonstrating qualification. Changes to training shall be implemented if indicated due to incidents potentially compromising safety, or if changes are made to facilities or processes.

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The training programs and maintenance of the training program records at the facility are the responsibility of the Training Manager. Accurate records are maintained on each employee's qualifications, experience, and training. The employee training file shall include records of all general employee training, technical training, and employee development training conducted at the facility. The employee training file shall also contain records of special company sponsored training conducted by others. The training records for each individual are maintained so that they are accurate and retrievable. Training records are retained in accordance with the records management system.

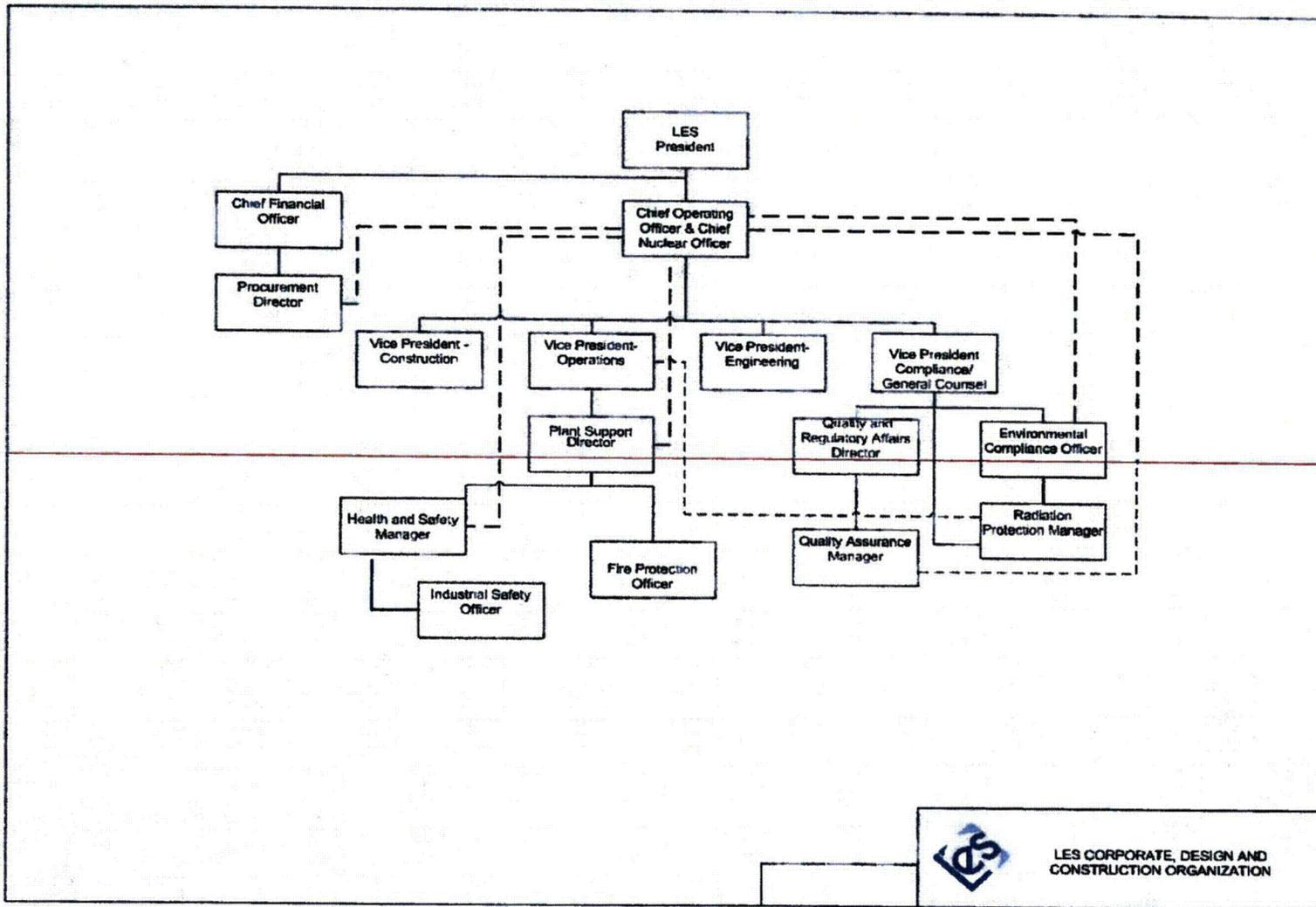
Additional details on the facility training program are provided in Chapter 11, Management Measures.

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.

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. Administrative procedures are written by each department as necessary to control activities that support process operations, including management measures (e.g. configuration management, training and record-keeping). Maintenance procedures address preventive and corrective maintenance, surveillance (includes calibration, inspection, and other surveillance testing), functional testing following maintenance, and requirements for pre-maintenance activity involving reviews of the work to be performed and reviews of procedures. Emergency procedures address the preplanned actions of operators and other plant personnel in the event of an emergency.

2.5 Chapter 2 Figures



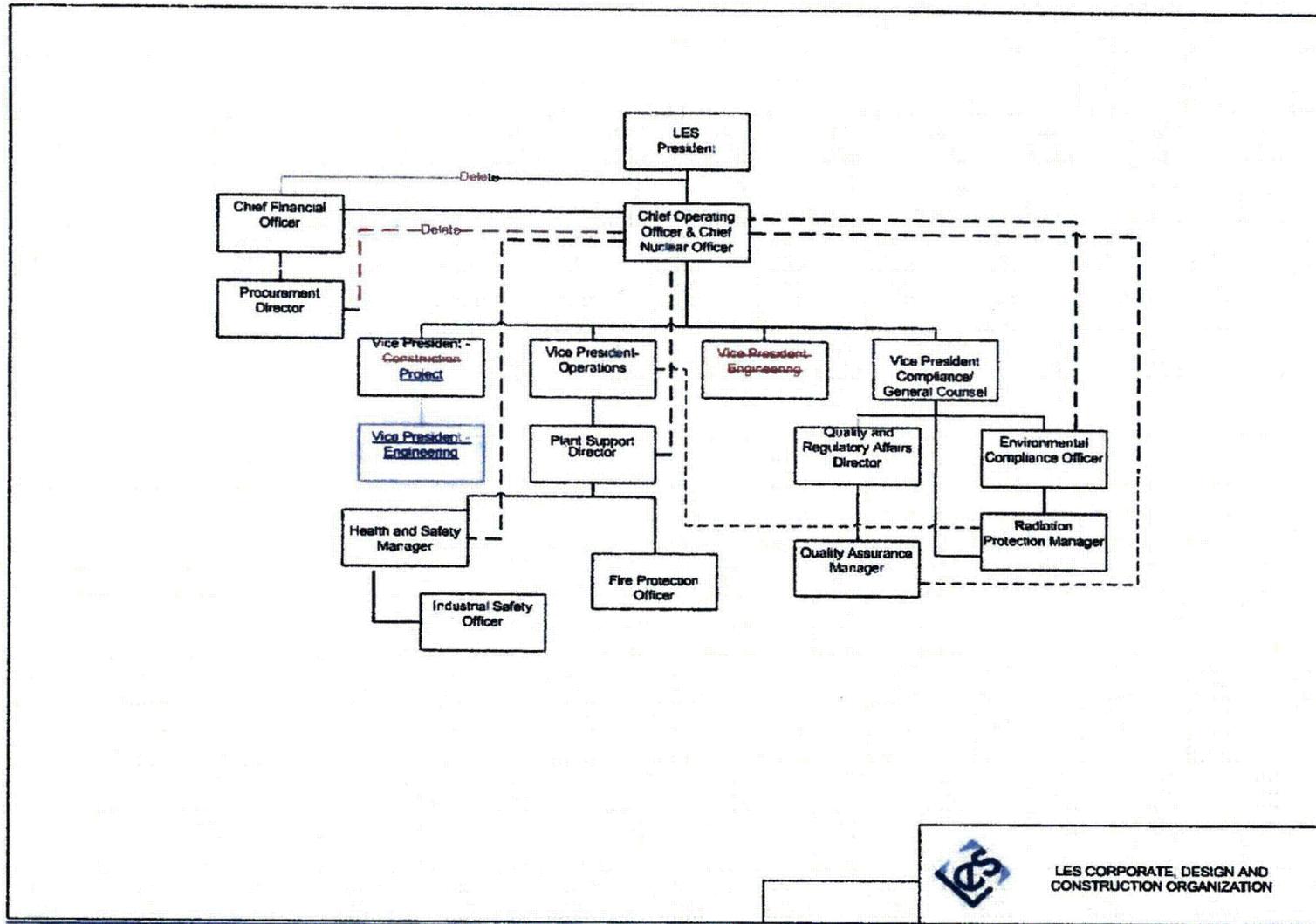
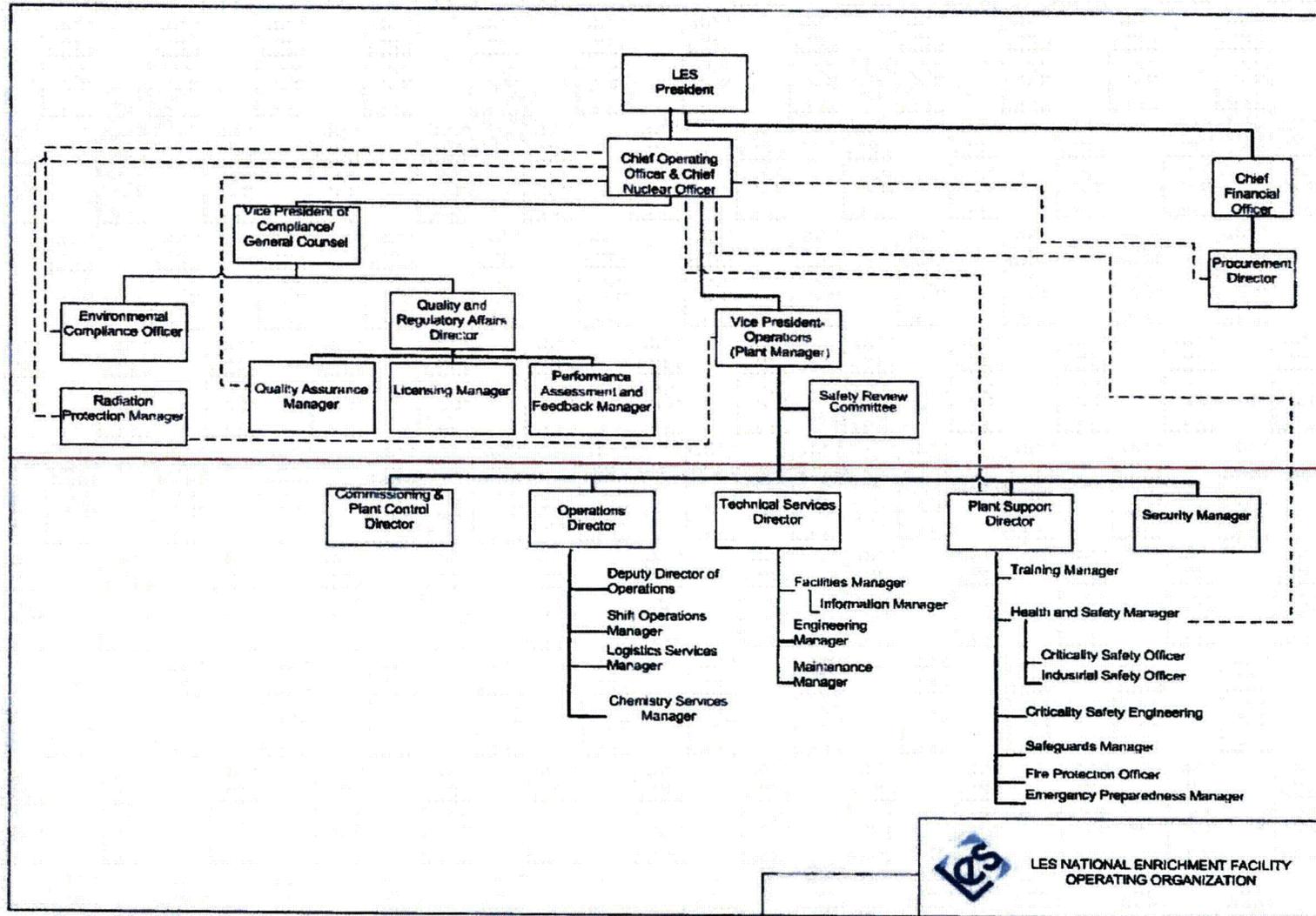


Figure 2.1-1 LES Corporate, Design and Construction Organization



2.5 Chapter 2 Figures



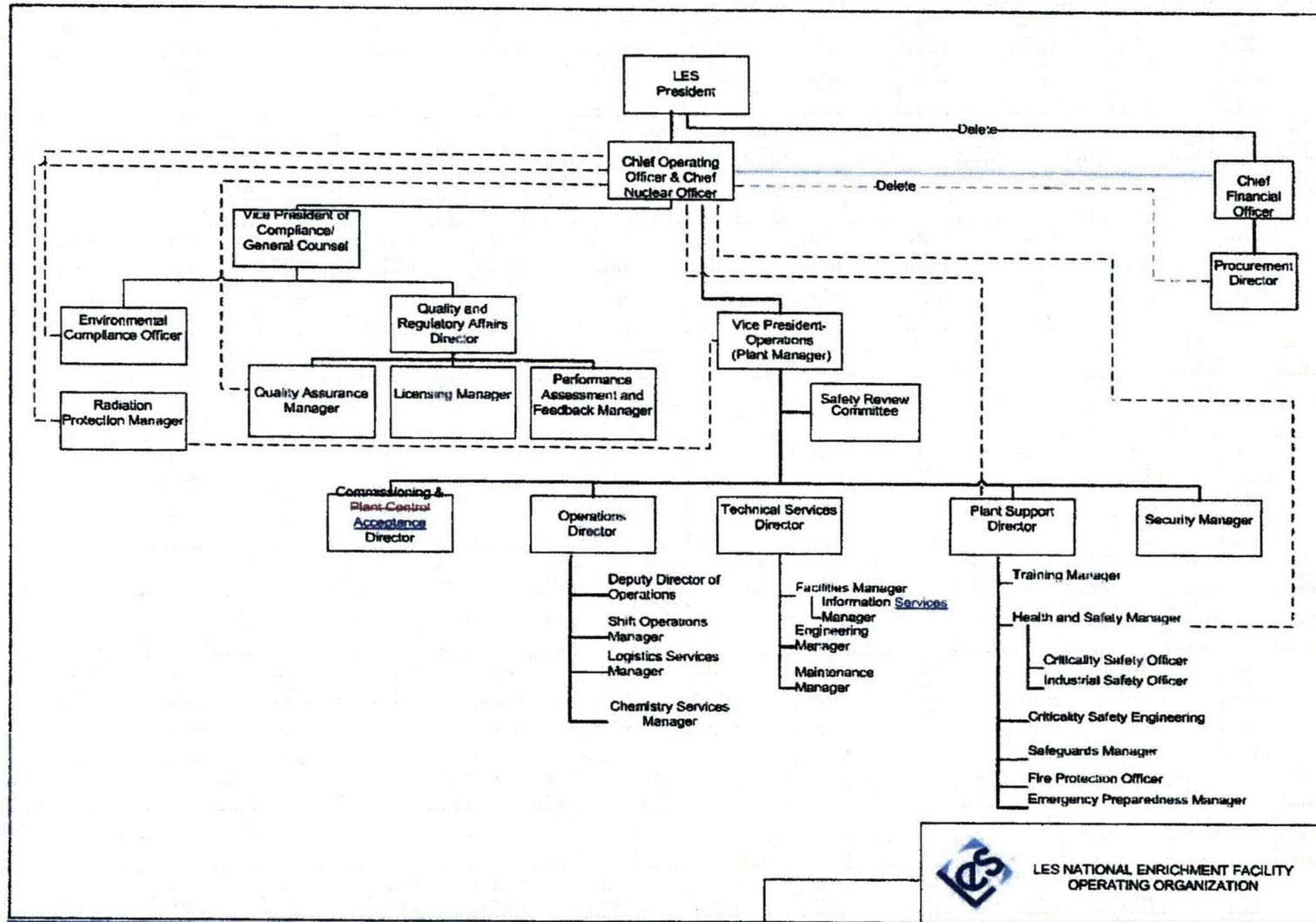


Figure 2.1-2 LES National Enrichment Facility Operating Organization

3.1 Safety Program

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.

Planned maintenance on engineered IROFS, or any items that may affect the function of IROFS, that do not have redundant functions available, 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.

For an IROFS that is found to be degraded or impaired by planned operations, maintenance, or construction activities: a compensatory measure may be used to ensure that the function of the IROFS is compensated until it is returned to service. For example, a continuous fire watch may be used to compensate for a degraded IROFS barrier.

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Corrective maintenance involves repair or replacement of equipment that has unexpectedly degraded or failed. Corrective maintenance restores the equipment to acceptable performance through a planned, systematic, controlled, and documented approach for the repair and replacement activities.

Following any maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the IROFS, as necessary, is performed to ensure the IROFS is capable of performing its intended safety function.

Training and Qualifications

IROFS, and any items that may affect the function of IROFS, require that personnel involved at each level (from design through and including any assumed process implementation steps or actions) have and maintain the appropriate training and qualifications. Employees are provided with training to establish the knowledge foundation and on-the-job training to develop work performance skills. For process implemented steps or actions, a needs/job analysis is performed and tasks are identified to ensure that appropriate training is provided to personnel working on tasks related to IROFS. Minimum training requirements are developed for those positions whose activities are related to IROFS. Initial identification of job-specific training requirements is based on experience. Entry-level criteria (e.g., education, technical background, and/or experience) for these positions are contained in position descriptions.

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 license or certification.

Continuing training is provided, as required, to maintain proficiency in specific knowledge and skill related activities. For all IROFS, and any items that may affect the function of IROFS, involving process implemented steps or actions, annual refresher training or requalification is generally required as identified in the needs/job analysis referenced in the previous paragraph. (any exceptions credited within the ISA are discussed in the National Enrichment Facility Integrated Safety Analysis Summary).

3.2 INTEGRATED SAFETY ANALYSIS METHODS

This section outlines the approach utilized for performing the integrated safety analysis (ISA) of the process accident sequences. The approach used for performing the ISA is consistent with Example Procedure for Accident Sequence Evaluation, Appendix A to Chapter 3 of NUREG-1520. This approach employs a semi-quantitative risk index method for categorizing accident sequences in terms of their likelihood of occurrence and their consequences of concern. The risk index method framework identifies which accident sequences have consequences that could exceed the performance requirements of

10 CFR 70.61 (CFR, 2003c) and, therefore, require designation of items relied on for safety (IROFS) and supporting management measures. Descriptions of these general types of higher consequence accident sequences are reported in the ISA Summary.

The ISA is a systematic analysis to identify plant and external hazards and the potential for initiating accident sequences, the potential accident sequences, the likelihood and consequences, and the IROFS.

The hazard and operability (HAZOP) analysis method was used initially to identify hazards for the Uranium Hexafluoride (UF₆) process systems and Technical Services Building (TSB) systems. This method is consistent with the guidance provided in NUREG-1513. The choice of a particular method or combination of methods is dependent upon a number of factors including:

- Analysis problem characteristics
- Motivation for the study
- Perceived risk associated with the subject process or activity
- Resource availability and analyst/management preference
- Type of information available to perform the study
- Type of results needed

To satisfy NRC requirements as defined in Part 70, a method should be chosen that is capable of identifying specific accident/event sequences in addition to the safety controls that prevent such accidents or mitigate their consequences. The HAZOP method has this capability.

NUREG-1513 identifies several methods in addition to the HAZOP method (i.e., What-If/Checklist and Failure Modes and Effects Analysis (FMEA)) that may be implemented. The guidance from NUREG-1513 will be followed for selection of a hazard analysis method.

~~The ISA uses a hazard analysis method to identify the hazards which are relevant for each system or facility.~~ The ISA Team reviewed the hazard identified for the "credible worst-case" consequences. All credible high or intermediate severity consequence accident scenarios were assigned accident sequence identifiers, accident sequence descriptions, and a risk index determination was made.

The risk index method is regarded as a screening method, not as a definitive method of proving the adequacy or inadequacy of the IROFS for any particular accident.

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3.2 Integrated Safety Analysis Methods

Some items that warrant special consideration during the integration process are:

- Common mode failures and common cause situations.
- Support system failures such as loss of electrical power or city water. Such failures can have a simultaneous effect on multiple systems.
- Divergent impacts of IROFS. Assurance must be provided that the negative impacts of an IROFS, if any, do not outweigh the positive impacts; i.e., to ensure that the application of an IROFS for one safety function does not degrade the defense-in-depth of an unrelated safety function.
- Other safety and mitigating factors that do not achieve the status of IROFS that could impact system performance.
- Identification of scenarios, events, or event sequences with multiple impacts, i.e. impacts on chemical safety, fire safety, criticality safety, and/or radiation safety. For example, a flood might cause both a loss of containment and moderation impacts.
- Potential interactions between processes, systems, areas, and buildings; any interdependence of systems, or potential transfer of energy or materials.
- Major hazards or events, which tend to be common cause situations leading to interactions between processes, systems, buildings, etc.

3.2.2 ~~Process~~ HAZOP Hazard Analysis Method

As noted above, the HAZOP method was used to identify the process hazards. The HAZOP process hazard analysis (PHA) method is consistent with the guidance provided in NUREG-1513. Implementation of the HAZOP method was accomplished by either validating the Urenco HAZOPs for the NEF design or performing a new HAZOP for systems where there were no existing HAZOPs. In general, new HAZOPs were performed for the Cylinder Receipt and Dispatch Building (CRDB) systems. In cases for which there was an existing HAZOP, the ISA Team, through the validation process, developed a new HAZOP.

For the UF₆ process systems, this portion of the ISA was a validation of the HAZOPs provided by Urenco. The validation process involved workshop meetings with the ISA Team. In the workshop meeting, the ISA Team challenged the results of the Urenco HAZOPs. As necessary the HAZOPs were revised/updated to be consistent with the requirements identified in

10 CFR 70 (CFR, 2003b) and as further described in NUREG-1513 and NUREG-1520.

To validate the Urenco HAZOPs, the ISA Team followed the HAZOP process as discussed in Guidelines for Hazard Evaluation Procedures (AIChE, 1992). Additional steps performed in this validation that are not identified in the above reference include: ~~performed the following tasks:~~

- ~~The Urenco process engineer described the salient points of the process system covered by the HAZOP being validated.~~
- ~~The ISA Team divided the process "Nodes" into reasonable functional blocks.~~
- ~~The process engineer described the salient points of the items covered by the "Node" being reviewed.~~

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3.2 Integrated Safety Analysis Methods

- ~~The ISA Team reviewed the "Guideword" used in the Urenco HAZOP to determine if the HAZOP is likely to identify all credible hazards. A representative list of the guidewords used by the ISA Team is provided in Table 3.1-1, HAZOP Guidewords, to ensure that a complete assessment was performed.~~
- ~~The ISA Team Leader introduced each Guideword being considered in the ISA HAZOP and the team reviewed and considered the potential hazards.~~
- The ISA Team created a list of deviations for the UF₆ process, other processes in which the deviation could potentially impact the UF₆ process, and for external events (i.e., deviations from normal weather or external activities).
- For each potential hazard, the ISA Team considered the causes, including potential interactions among materials. Then, for each cause, the ISA Team considered the consequences and consequence severity category for the consequences of interest (Criticality Events, Chemical Releases, Radiation Exposure, Environment impacts). A statement of "No Safety Issue" was noted in the system HAZOP table for consequences of no interest such as maintenance problems or industrial personnel accidents.
- ~~For each hazard, the ISA Team considered existing safeguards designed to prevent the hazard from occurring.~~
- For each hazard, in addition to identification of safeguards, the ISA Team also considered any existing design features that could mitigate/reduce the consequences.
- ~~The Urenco HAZOP was modified to reflect the ISA Team's input in the areas of hazards, causes, consequences, safeguards and mitigating features.~~
- For each external event hazard, the ISA Team determined if the external hazard is credible (i.e., external event initiating frequency >10⁻⁶ per year).
- ~~When all of the Guidewords had been considered for a particular node, the ISA Team applied the same process and guidewords to the next node until the entire process system was completed.~~

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The Urenco HAZOP was modified to reflect the ISA Team's input in the areas of hazards, causes, consequences, safeguards and mitigating features.

The same process as above was followed for the CRDB systems, except that instead of using the validation process, the ISA Team developed a completely new HAZOP. This HAZOP was then used as the hazard identification input into the remainder of the process.

The results of the ISA Team workshops are summarized in the ISA HAZOP Table, which forms the basis of the hazards portion of the Hazard and Risk Determination Analysis. The HAZOP tables are contained in the ISA documentation. The format for this table, which has spaces for describing the node under consideration and the date of the workshop, is provided in Table 3.1-2, ISA HAZOP Table Sample Format. This table is divided into 7 columns:

GUIDEWORD	Identifies the Guideword under consideration.
HAZARD	Identifies any issues that are raised.
CAUSES	Lists any and all causes of the hazard noted.

3.2 Integrated Safety Analysis Methods

CONSEQUENCES	Identifies the potential and worst case consequence and consequences severity category if the hazard goes uncontrolled.
SAFEGUARDS	Identifies the engineered and/or administrative protection designed to prevent the hazard from occurring.
MITIGATION	Identifies any protection, engineered or otherwise, that can mitigate/reduce the consequences.
COMMENTS	Notes any comments and any actions requiring resolution.

This approach was used for all of the process system hazard identifications. The "Fire" and "External Events" guidewords were handled as a facility-wide assessment and were not explicitly covered in each system hazard evaluation.

The results of the HAZOP are used directly as input to the risk matrix development.

3.2.3 What-If/Checklist Hazard Analysis Method

The guidance from NUREG-1513 is followed for the What-If/Checklist hazard analysis method selection. The What-If/Checklist Analysis technique is a combination of two hazard evaluation methods: What-If Analysis and Checklist Analysis. The method is performed by an ISA Team with personnel experienced with the subject process. The ISA Team uses the What-If Analysis technique to brainstorm various types of process accidents that can occur. Then the ISA Team uses one or more checklists to help fill in any gaps that may have been missed. Rather than focusing on a specific list of design or operating features, checklists used in a What-If/Checklist Analysis are more general and focus on sources of hazards and accidents.

A What-If/Checklist Analysis consists of the following steps: (1) preparing for the review, (2) developing a list of What-If questions and issues, (3) using a checklist to cover any gaps, (4) evaluating each of the questions and issues, and (5) documenting the results.

For each What-If question, the ISA Team determines the likelihood, consequences, safeguards, and acceptability of risk. The ISA Team meetings results are summarized in the What-If/Checklist, which forms the Hazard and Risk Determination Analysis basis.

3.2.4 Failure Modes and Effects Analysis (FMEA) Hazard Analysis Method

The guidance from NUREG-1513 recommends the FMEA hazard analysis method use. The FMEA is a systematic method for examining the effects of component failures on system performance. To perform the FMEA, an individual analyst lists all the components in the system under review, as well as all the failure modes for these components. The ISA Team made of analysts familiar with the system then identifies the hazards associated with each component failure and suggests corrective actions when appropriate.

The FMAE technique:

- Defines physical system bounds
- Determines the effect of each component failure mode

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- Identifies safeguards to protect against the causes and/or consequences of each component failure mode
- Lists system components and postulates failure mode for each component and each physical bound
- Suggests actions for improving the system if the risk is deemed unacceptable

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3.2.33.2.5 Risk Matrix Development

3.2.3.43.2.5.1 Consequence Analysis Method

10 CFR 70.61 (CFR, 2003c) specifies two categories for accident sequence consequences: "high consequences" and "intermediate consequences." Implicitly there is a third category for accidents that produce consequences less than "intermediate." These are referred to as "low consequence" accident sequences. The primary purpose of PHA is to identify all uncontrolled and unmitigated accident sequences. These accident sequences are then categorized into one of the three consequence categories (high, intermediate, low) based on their forecast radiological, chemical, and/or environmental impacts.

For evaluating the magnitude of the accident consequences, calculations were performed using the methodology described in the ISA documentation. Because the consequences of concern are the chemotoxic exposure to HF and UO₂F₂, the dispersion methodology discussed in Section 6.3.2 was used. The dose consequences for all of the accident sequences were evaluated and compared to the criteria for "high" and "intermediate" consequences. The inventory of uranic material for each accident considered was dependent on the specific accident sequence. For criticality accidents, the consequences were conservatively assumed to be high for both the public and workers.

Table 3.1-3, Consequence Severity Categories Based on 10 CFR 70.61, presents the radiological and chemical consequence severity limits of 10 CFR 70.61 (CFR, 2003c) for each of the three accident consequence categories. Table 3.1-4, Chemical Dose Information, provides information on the chemical dose limits specific to the NEF.

3.2.3.23.2.5.2 Likelihood Evaluation Method

10 CFR 70.61 (CFR, 2003c) also specifies the permissible likelihood of occurrence of accident sequences of different consequences. "High consequence" accident sequences must be "highly unlikely" and "intermediate consequence" accident sequences must be "unlikely." Implicitly, accidents in the "low consequence" category can have a likelihood of occurrence less than "unlikely" or simply "not unlikely." Table 3.1-5, Likelihood Categories Based on 10 CFR 70.61, shows the likelihood of occurrence limits of 10 CFR 70.61 (CFR, 2003c) for each of the three likelihood categories.

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The definitions of "not unlikely" and "unlikely" are taken from NUREG-1520. The definition of "highly unlikely" is taken from NUREG-1520. Additionally, a qualitative determination of "highly unlikely" can apply to passive design component features (e.g., tanks, piping, cylinders, etc.) of the facility that do not rely on human interface to perform the criticality safety function (i.e., termed "safe-by-design"). Safe-by-design components are those components that by their physical size or arrangement have been shown to have a $k_{\text{eff}} < 0.95$. The definition of safe-by-design components encompasses two different categories of components. The first category includes those components that are safe-by-volume, safe-by-diameter or safe-by-slab

3.2 Integrated Safety Analysis Methods

- b. A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive (In determining that there is no reason for such actions, a wide range of possible motives, short of intent to cause harm, must be considered. Necessarily, no such sequence of events can ever have actually happened in any fuel cycle facility.)
- c. Process deviations for which there is a convincing argument, given physical laws that they are not possible, or are unquestionably extremely unlikely.

3.2.3.33.2.5.3 Risk Matrix

The three categories of consequence and likelihood can be displayed as a 3 x 3 risk index matrix. By assigning a number to each category of consequence and likelihood, a qualitative risk index can be calculated for each combination of consequence and likelihood. The risk index equals the product of the integers assigned to the respective consequence and likelihood categories. The risk index matrix, along with computed risk index values, is illustrated in Table 3.1-6, Risk Matrix with Risk Index Values. The shaded blocks identify accidents of which the consequences and likelihoods yield an unacceptable risk index and for which IROFS must be applied.

The risk indices can initially be used to examine whether the consequences of an uncontrolled and unmitigated accident sequence (i.e., without any IROFS) could exceed the performance requirements of 10 CFR 70.61 (CFR, 2003c). If the performance requirements could be exceeded, IROFS are designated to prevent the accident or to mitigate its consequences to an acceptable level. A risk index value less than or equal to four means the accident sequence is acceptably protected and/or mitigated. If the risk index of an uncontrolled and unmitigated accident sequence exceeds four, the likelihood of the accident must be reduced through designation of IROFS. In this risk index method, the likelihood index for the uncontrolled and unmitigated accident sequence is adjusted by adding a score corresponding to the type and number of IROFS that have been designated.

3.2.43.2.6 Risk Index Evaluation Summary

The results of the ISA are summarized in tabular form. This table includes the accident sequences identified for this facility. The accident sequences were not grouped as a single accident type but instead were listed individually in the table. The Table has columns for the initiating event and for IROFS. IROFS may be mitigative or preventive. Mitigative IROFS are measures that reduce the consequences of an accident. The phrase "uncontrolled and/or unmitigated consequences" describes the results when the system of existing preventive IROFS fails and existing mitigation also fails. Mitigated consequences result when the preventive IROFS fail, but mitigative measures succeed. Index numbers are assigned to initiating events, IROFS failure events, and mitigation failure events, based on the reliability characteristics of these items.

With redundant IROFS and in certain other cases, there are sequences in which an initiating event places the system in a vulnerable state. While the system is in this vulnerable state, an IROFS must fail for the accident to result. Thus, the frequency of the accident depends on the frequency of the first event, the duration of vulnerability, and the frequency of the second IROFS failure. For this reason, the duration of the vulnerable state is considered, and a duration index is assigned. The values of all index numbers for a sequence, depending on the number of events involved, are added to obtain a total likelihood index, T. Accident sequences are then

3.6 CHAPTER 3 TABLES

Table 3.1-1 HAZOP Guidewords

UF₆ PROCESS GUIDEWORDS			
Less Heat	Corrosion	Maintenance	No Flow
More Heat	Loss of Services	Criticality	Reverse Flow
Less Pressure	Toxicity	Effluents/Waste	Less Uranium
More Pressure	Contamination	Internal Missile	More Uranium
Impact/Drop	Loss of Containment	Less Flow	Light Gas
Fire (Process, internal, other)	Radiation	More Flow	External Event
NON-UF₆ PROCESS GUIDEWORDS			
High Flow	Low Pressure	Impact/Drop	More Uranium
Low Flow	High Temperature	Corrosion	External Event
No Flow	Low Temperature	Loss of Services	Startup
Reverse Flow	Fire	Toxicity	Shutdown
High Level	High Contamination	Radiation	Internal Missile
Low Level	Rupture	Maintenance	
High Pressure	Loss of Containment	Criticality	
No Flow			
EXTERNAL EVENTS POTENTIAL CAUSES			
Construction on Site	Hurricane	Seismic	Transport Hazard Off-Site
Flooding	Industrial Hazard Off-site	Tornado	External Fire
Airplane	Snow/Ice	Local Intense Precipitation	

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4.1 Commitment to Radiation Protection Program Implementation

The facility corrective action process is implemented if (1) personnel dose monitoring results or ~~personnel contamination levels~~ exceed the administrative personnel limits; or if an incident results in airborne occupational exposures exceeding the administrative limits or (2) the dose limits in 10 CFR 20 (CFR, 2003b), Appendix B or 10 CFR 70.61 (CFR, 2003e) are exceeded.

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The information developed from the corrective action process is used to improve radiation protection practices and to preclude the recurrence of similar incidents. If an incident as described in item two above occurs, the NRC is informed of the corrective action taken or planned to prevent recurrence and the schedule established by the facility to achieve full compliance. The corrective action process and incident investigation process are described in Section 11.6, Incident Investigations and Corrective Action Process.

4.1.1 Responsibilities of Key Program Personnel

This section describes the Radiation Protection Program's organizational structure and the responsibilities of key personnel are discussed. These personnel play an important role in the protection of workers, the environment and implementation of the ALARA program. Chapter 2, Organization and Administration, discusses the facility organization and administration in further detail. Section 2.2, Key Management Positions of Chapter 2, presents a detailed discussion of the responsibilities of key management personnel.

4.1.1.1 Plant Manager

The Plant Manager is responsible for all aspects of facility operation, including the protection of all persons against radiation exposure resulting from facility operations and materials, and for compliance with applicable NRC regulations and the facility license.

4.1.1.2 Chemistry Services Manager

The Chemistry Services Manager reports to the Operations Director and has the responsibility for directing the activities that ensure the facility maintains compliance with appropriate rules, regulations, and codes. This includes activities associated with nuclear safety. The Chemistry Services Manager works with the other facility managers to ensure consistent interpretations of HS&E requirements performs independent reviews and supports facility and operations change control reviews.

4.1.1.3 Radiation Protection Manager

The Radiation Protection Manager reports to the Vice President Compliance/General Counsel and is responsible for implementing the Radiation Protection Program. In matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager. The Radiation Protection Manager has a line of communication with the Plant Support Manager to ensure objective radiation protection audit, review and control activities are maintained.

The Radiation Protection Manager is responsible for:

- Establishing the Radiation Protection Program
- Generating and maintaining procedures associated with the program

4.1 Commitment to Radiation Protection Program Implementation

- Reviewing and auditing the efficacy of the program in complying with NRC and other governmental regulations and applicable Regulatory Guides
- Modifying the program based upon experience and facility history
- Adequately staffing the Radiation Protection group to implement the Radiation Protection Program
- Establishing and maintaining an ALARA program and assuring it is practiced by all personnel
- Establishing and maintaining a respirator usage program
- Monitoring worker doses, both internal and external
- Complying with the radioactive materials possession limits for the facility
- Handling of radioactive wastes for disposal
- Calibration and quality assurance of all radiological instrumentation, including verification of required Lower Limits of Detection or alarm levels
- Establishing and maintaining a radiation safety training program for personnel working in Restricted Areas and any Radiologically Controlled Area (RCA)
- Performing audits of the Radiation Protection Program on an annual basis
- Establishing and maintaining the radiological environmental monitoring program
- Posting ~~the Restricted Areas~~ in any RCA, and within these areas, posting: Radiation, Airborne Radioactivity, High Radiation and Contaminated Areas as appropriate; and developing occupancy guidelines for these areas as needed.

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4.1.1.4 Shift Operations Manager

The Shift Operations Manager is responsible for operating the facility safely and in accordance with procedures so that any effluents released to the environment and all exposures to the public and facility personnel are within the limits specified in applicable regulations, procedures and guidance documents.

4.1.1.5 Facility Personnel

Facility personnel are required to work safely and to follow the rules, regulations and procedures that have been established for their protection and the protection of the public. Personnel whose duties require (1) working with radioactive material, (2) entering radiation areas, (3) controlling facility operations that could affect effluent releases, or (4) directing the activities of others, are trained such that they understand and effectively carry out their responsibilities.

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, Selection, Qualification and Training of Personnel for Nuclear Power Plants.

Radiation Protection Program resources in terms of staffing and equipment are provided to implement an effective Radiation Protection Program and response to emergencies in

4.2 COMMITMENT TO AN ALARA PROGRAM

Section 4.1, Commitment to Radiation Protection Program Implementation, above states the facility's commitment to the implementation of an ALARA program. The objective of the program is to make every reasonable effort to maintain facility exposures to radiation as far below the dose limits of 10 CFR 20.1201 (CFR, 2003f) as is practical and to maintain radiation exposures to members of the public such that they are not expected to receive the dose limits of 10 CFR 20.1101(d) (CFR, 2003d). The design and implementation of the ALARA program is consistent with the guidance provided in Regulatory Guides 8.2, 8.13, 8.29, and 8.37. The operation of the facility is consistent with the guidance provided in Regulatory Guide 8.10.

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 (CFR, 2003g).

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 (CFR, 2003d) limits. As discussed in Section 4.7, Radiation Surveys and Monitoring Programs Commitments, ~~radiological zones~~ RCA's or Restricted Areas designated as RCAs are established within the facility to support the ALARA commitment by minimizing the spread of contamination and reduce unnecessary exposure of personnel to radiation.

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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 by providing adequate space for ease of maintenance in areas with higher dose rates, reducing the length of time required to complete the task, thereby reducing the time of exposure. Areas where facility personnel spend significant amounts of time are designed to maintain the lowest dose rates reasonably achievable.

The Radiation Protection Manager is responsible for implementing the ALARA program and ensuring that adequate resources are committed to make the program effective. The Radiation Protection Manager prepares an annual ALARA program evaluation report. The report reviews (1) radiological exposure and effluent release data for trends, (2) audits and inspections, (3) use, maintenance and surveillance of equipment used for exposure and effluent control, and (4) other issues, as appropriate, that may influence the effectiveness of the radiation protection/ALARA programs. Copies of the report are submitted to the Plant Manager, Radiation Safety Committee, and the Safety Review Committee.

4.2.1 ALARA Committee

The Safety Review Committee (SRC) fulfills the duties of the ALARA Committee and meets at least quarterly. Additional details concerning the membership and qualifications of the SRC are provided in Chapter 2, Organization and Administration.

4.4 Commitment to Written Procedures

4.4 COMMITMENT TO WRITTEN PROCEDURES

All operations at LES involving licensed materials are conducted through the use of procedures as required by 10 CFR 70.22(8) (CFR, 2003h). Radiation protection procedures are prepared, reviewed and approved to carry out activities related to the radiation protection program. Procedures are used to control radiation protection activities to ensure that the activities are carried out in a safe, effective and consistent manner. Radiation protection procedures are reviewed and revised as necessary, to incorporate any facility or operational changes or changes to the facility's Integrated Safety Analysis (ISA) in the License Basis Documents.

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The radiation protection procedures are assigned to qualified personnel. Initial procedure drafts are reviewed by members of the facility staff and other personnel with enrichment plant operating experience. The Radiation Protection Manager (or a designee who has the qualifications of the Radiation Protection Manager) reviews and approves procedures as well as proposed revisions to procedures.

4.4.1 Radiation Work Permits Procedures

All work performed in a radiological area Radiologically Controlled Area (RCA) is performed in accordance with a Radiation Work Permit (RWP). The procedures controlling RWPs are consistent with the guidance provided in Regulatory Guide 8.10. An RWP may also be required whenever the Radiation Protection Manager deems that one is necessary. Activities involving licensed materials not covered by operating procedures and where direct or airborne radioactivity levels are likely to exceed airborne radioactivity administrative or regulatory limits require the issuance of a RWP. Both routine and non-routine activities are performed under a RWP. The RWP provides a description of the authorized activities and 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 Radiation Protection Manager or designee. The designee must meet the requirements of Section 4.1.2, Staffing of the Radiation Protection Program. RWPs have a predetermined period of validity with a specified expiration or termination time.

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Standing RWPs are issued for routinely performed activities in areas where radiological conditions are well characterized and not expected to change, such as tours of the plant by shift personnel or the charging of cylinders. A Standing RWP would, for example, be used for the job evolution of cylinder charging; a new RWP is not issued each time a new cylinder is charged.

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Listed below are requirements of the RWP procedures.

- The 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 RCAs or work with licensed materials are reviewed by the Radiation Protection Manager or designee for the potential to cause radiation exposures to exceed action levels or to produce radioactive contamination

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4.4 Commitment to Written Procedures

- RWPs 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 | LBDCR-10-0042
- RWPs are posted at access points to ~~Restricted Areas with copies of current RWPs posted at the work area location~~ RCAs | LBDCR-10-0042
- RWPs 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
- RWPs are retained as a record ~~at least for the life of the facility~~ until termination of the license requiring the record in compliance with 10 CFR 20.2103 (CFR, 2003v). | LBDCR-10-0042

4.5 Training Commitments

4.5 TRAINING COMMITMENTS

The design and implementation of the radiation protection training program complies with the requirements of 10 CFR 19.12 (CFR, 2003i). Records are maintained in accordance with 10 CFR 20.2110 (CFR, 2003j).

The development and implementation of the radiation protection training program is consistent with the training development guidance provided in the following regulatory guidance documents:

- Regulatory Guide 8.10-Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable
- Regulatory Guide 8.13-Instructions Concerning Prenatal Radiation Exposure
- Regulatory Guide 8.29-Instructions Concerning Risks From Occupational Radiation Exposure
- ASTM E1168-Radiological Protection Training for Nuclear Facility Workers.

All personnel and visitors entering the Restricted Areas or Radiologically Controlled Areas (RCAs) receive training that is commensurate with the radiological hazard to which they may be exposed. Alternatively, visitors will be provided with trained escorts who have received radiation protection training for entry into the RCA.

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The level of radiation protection training is based on the potential radiological health risks associated with an employee's work responsibilities. In accordance with provisions of 10 CFR 19.12 (CFR, 2003i) any individual working at the facility that is likely to receive in a year a dose in excess of 1 mSv (100 mrem) is:

- A. Kept informed of the storage, transfer, or use of radioactive material
- B. Instructed in the health protection problems associated with exposure to radiation and radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed
- C. Required to observe, to the extent within the worker's control, the applicable provisions of the NRC regulations and licenses for the protection of personnel from exposure to radiation and radioactive material
- D. Instructed of their responsibility to report promptly to the facility management, any condition which may cause a violation of NRC regulations and licenses or unnecessary exposure to radiation and radioactive material
- E. Instructed in the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation and radioactive material
- F. Advised of the various notifications and reports to individuals that a worker may request in accordance with 10 CFR 19.13 (CFR, 2003k).

4.5 Training Commitments

The radiation protection training program takes into consideration a worker's 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 work place.

Continuing Training of personnel with access to the Restricted Area ~~previously trained~~ is performed for radiological, chemical, industrial, and criticality safety at least annually. The continuing training program also provides information on position specific/related procedure changes as appropriate and updating and changes in required skills. Changes to training are implemented, as necessary due to any incidents potentially compromising safety or if changes are made to the facility or processes. Training Records are maintained in accordance with LES records management system. Training programs are established in accordance with Section 11.3, Training and Qualifications. The radiation protection training program is evaluated at least annually to ensure it remains current and adequate to assure worker safety.

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The specifics of the Radiation Protection Training are described in the following section.

4.5.1 Radiation Protection Training

Radiation protection training emphasizes 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., operator, maintenance radiation protection technician, contractor personnel) commensurate with the radiation safety responsibilities associated with each position. ~~Visitors to a Restricted Area are trained in the formal training program or who have not completed nuclear safety training~~ are escorted by trained personnel while in the ~~Restricted Area~~ an RCA. Visitor to the RCA receive a radiological briefing commensurate with their entry in accordance with 10 CFR 19.12.

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Personnel access procedures ensure the completion of nuclear safety training prior to permitting unescorted access into the ~~Restricted Area~~ an RCA. Training sessions covering criticality safety, radiation protection and emergency procedures are conducted on a regular basis to accommodate new employees or those requiring continuing training. Continuing training is conducted when necessary to address changes in policies, procedures, requirements and the ISA.

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Specific topics covered in the training program are listed in Chapter 11, Management Measures, Section 11.3.3.1.1. The training provided includes the requirements of 10 CFR 19 (CFR, 2003a).

Individuals attending these sessions must pass an initial examination covering the training contents to assure the understanding and effectiveness of the training. The effectiveness of the training programs is also evaluated by audits and assessments of operations and maintenance personnel responsible for following the requirements related to the topics listed above.

Since contractor employees may perform diverse tasks in the ~~Restricted Areas or Radiologically Controlled Areas (RCAs)~~ of the facility, 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. The Radiation Protection Manager is responsible for ~~establishing and maintaining~~ approving the radiation protection training for all personnel, including contractor

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4.5 Training Commitments

personnel working at the facility. Records are maintained for each employee documenting the training date, scope of the training, identity of the trainer(s), any test results and other associated information by the Training Manager.

~~Individuals requiring unescorted access to a Restricted Area receive annual continuing training.~~ Contents of the radiation protection program is reviewed and updated through curriculum meetings at least every two years by the Radiation Protection Manager to ensure that the programs are current and adequate.

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4.6 VENTILATION AND RESPIRATORY PROTECTION PROGRAMS COMMITMENTS

The regulations contained in 10 CFR 20 (CFR, 2003b), 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.

The design of the ventilation and respiratory protection programs is consistent with the guidance contained in the following documents:

- Regulatory Guide 8.24-Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication
- ANSI N510-Testing of Nuclear Air Cleaning Systems
- ERDA 76-21-Nuclear Air Cleaning Handbook
- NCRP Report No. 59127-Operational Radiation Safety Program
- Regulatory Guide 8.15-Acceptable Programs for Respiratory Protection
- ~~ANSI Z88.2-Practices for Respiratory Protection~~

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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 UF₆ within process equipment. The entire UF₆ enrichment process, except for liquid sampling, is operated under a partial vacuum so that leaks are into the system and not into work areas.

Building ventilation systems control the temperature and the humidity of the air inside the building. Note: Not all buildings will have humidity control. ~~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 and exhaust 100% of the air handled to the environment through the exhaust stacks. All air released from potentially contaminated areas is filtered to remove radioactive particulates before it is released. The ventilation systems for potentially contaminated areas are designed to maintain the potentially contaminated areas at a slightly negative pressure relative to the uncontaminated areas. This ensures that the airflow direction is from areas of little or no contamination to areas of higher contamination.

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4.6 Ventilation and Respiratory Protection Programs Commitments

Process vents from the SBMs are collected by the Pumped Extract GEVS. Process vents in the CRDB (including fume hoods) are collected by the CRDB GEVS and by the Confinement Ventilation function of HVAC system. Air released from the Centrifuge Test Facility and the Centrifuge Post Mortem Facilities is filtered by the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System prior to release. The systems operate slightly below atmospheric pressure to remove potentially hazardous vapors and particulate from confined areas of the plant. The systems contain particulate and carbon adsorption filters to remove radioactive materials from the gas stream prior to release from the plant. GEVS have continuous HF monitors upstream and downstream of the filters and in the exhaust stack with high level alarms to inform operators of UF_6 releases in the plant. In the Centrifuge Test and Post Mortem Facility exhaust filtration system, a continuous HF monitor is provided in the exhaust stack.

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 since 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 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 set points 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.

~~Gloveboxes are designed to maintain a negative differential pressure of about 0.623 mbar (0.25 in H_2O) and is maintained anytime that the glovebox is in use. If the differential pressure is lost, use of the glovebox is suspended until the required differential pressure is restored.~~

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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, Corrective Maintenance and 11.2.3, Preventive Maintenance respectively.

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:

4.6 Ventilation and Respiratory Protection Programs Commitments

performed before the first field use of tight fitting, face-sealing respirators. Subsequent testing is performed at least annually thereafter. Fit testing must be performed with the facepiece operating in the negative pressure mode.

1. Each user is informed that they may leave the area at any time for relief from respirator use in the event of equipment malfunction, physical or psychological distress, procedural or communication failure, significant deterioration of operating conditions, or any other conditions that might require such relief.
2. In the selection and use of respirators, the facility provides for vision correction, adequate communication, low temperature work environments, and the concurrent use of other safety or radiological protection equipment. Radiological protection equipment is used in such a way as not to interfere with the proper operation of the respirator.
3. Standby rescue persons are used whenever one-piece atmosphere-supplying suits are in use. Standby rescue personnel are also used when any combination of supplied air respiratory protection device and personnel protective equipment is in use that presents difficulty for the wearer to remove the equipment. The standby personnel are equipped with respiratory protection devices or other apparatus appropriate for the potential hazards. The standby rescue personnel observe and maintain continuous communication with the workers (visual, voice, signal line, telephone, radio, or other suitable means). The rescue personnel are immediately available to assist the workers in case of a failure of the air supply or for any other emergency. The Radiation Protection Manager, in consultation with the Industrial Safety Officer, specifies the number of standby rescue personnel that must be immediately available to assist all users of this type of equipment and to provide effective emergency rescue if needed.
4. If Atmosphere-supplying respirators are used, they must be supplied with respirable air of grade D quality or better as defined by the Compressed Gas Association in publication G-7.1, Commodity Specification for Air and included in the regulations of the Occupational Safety and Health Administration (29 CFR 1910.134(i)(1)(ii)(A) through (E) (CFR, 2003I)).
5. No objects, materials or substances (such as facial hair), or any conditions that interfere with the face-to-facepiece seal or valve function, and that are under the control of the respirator wearer, are allowed between the skin of the wearer's face and the sealing surface of a tight-fitting respirator facepiece.

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The dose to individuals from the intake of airborne radioactive material is estimated by dividing the ambient air concentration outside the respirator by the assigned protection factor. If the actual dose is later found to be greater than that estimated initially, the corrected value is used. If the dose is later found to be less than the estimated dose, the lower corrected value may be used.

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 Section 11.7, Records Management. Respiratory protection procedures are revised as necessary whenever changes are made to the facility, processing or equipment.

4.7 RADIATION SURVEYS AND MONITORING PROGRAMS COMMITMENTS

Radiation surveys are conducted for two purposes: (1) to ascertain radiation levels, concentrations of radioactive materials, and potential radiological hazards that could be present in the facility; and (2) to detect releases of radioactive material from facility equipment and operations. Radiation surveys focus on those areas of the facility identified in the ISA where the occupational radiation dose limits could potentially be exceeded. Measurements of airborne radioactive material and/or bioassays are used to determine that internal occupational exposures to radiation do not exceed the dose limits specified in 10 CFR 20 (CFR, 2003b), Subpart C.

Written procedures for the radiation survey and monitoring programs assure compliance with the requirements of 10 CFR 20 (CFR, 2003b) Subpart F (Surveys and Monitoring), Subpart C (Occupational Dose Limits), Subpart L (Records) and Subpart M (Reports).

The radiation survey and monitoring programs are consistent with the guidance provided in the following references:

- Regulatory Guide 8.2-Guide for Administrative Practice in Radiation Monitoring
- Regulatory Guide 8.13-Instructions Concerning Prenatal Radiation Exposure
- Regulatory Guide 8.28-Audible Alarm Dosimeters
- Regulatory Guide 8.36-Radiation Protection to the Embryo/Fetus
- Regulatory Guide 8.4-Direct-Reading and Indirect-Reading Pocket Dosimeters
- Regulatory Guide 8.7- Instructions for Recording and Reporting Occupational Radiation Exposure Data
- Regulatory Guide 8.9-Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program
- Regulatory Guide 8.24-Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication
- Regulatory Guide 8.25-Air Sampling in the Workplace
- Regulatory Guide 8.30-Health Physics Surveys in Uranium Recovery Facilities
- Regulatory Guide 8.34-Monitoring Criteria and Methods To Calculate Occupational Radiation Doses
- NUREG-1400-Air Sampling in the Workplace
- ANSI/HPS N13.1-Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities
- ANSI N323-Radiation Protection Instrumentation Test and Calibration
- ANSI N13.11-Dosimetry-Personnel Dosimetry Performance-Criteria for Testing
- ANSI N13.15-Radiation Detectors-Personnel Thermoluminescence Dosimetry Systems-Performance
- ANSI/HPS N13.22-Bioassay Program for Uranium

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- ~~ANSI N13.27 Performance Requirements for Pocket-Sized Alarm Dosimeters and Alarm Ratemeters~~
- ANSI/HPS N13.30-Performance Criteria for Radiobioassay
- ANSI N13.6, Practice for Occupational Radiation Exposure Records Systems

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Facility procedures include an outline of the program objectives, sampling procedures and data analysis methods. Equipment selection is based on the type of radiation being monitored. Procedures are prepared for each of the instruments used and specify the frequency and method of calibration. Maintenance and calibration are in accordance with the manufacturers' recommendations. Specific types of instruments used in the facility are discussed below.

The survey program procedures specify the frequency of measurements and record keeping and reporting requirements. As stated in Section 4.1, Commitment to Radiation Protection Program Implementation, the facility corrective action process is implemented if: 1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; or if an incident results in airborne occupational exposures exceeding the administrative limits, or 2) the dose limits in 10 CFR 20, Appendix B (CFR, 2003m) or 10 CFR 70.61 (CFR, 2003e) are exceeded. In the event the occupational dose limits given in 10 CFR 20 (CFR, 2003b), Subpart C are exceeded, notification of the NRC is in accordance with the requirements of 10 CFR 20, Subpart M—Reports.

All personnel who enter ~~Restricted Areas~~ RCA (defined in Section 4.7.1.23) 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 for contamination prior to exiting ~~Restricted Areas~~ RCA.

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Continuous airborne radioactivity monitors provide indication of the airborne activity levels in the ~~Restricted Areas~~ RCAs of the facility. Monitoring instruments for airborne alpha emitters are provided at different locations throughout 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 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 and decay chains for these radionuclides are shown in Table 4.7-1, Radiation Emitted from Natural UF₆ Feed, and Figure 4.7-1, Uranium and Decay Products of Interest, respectively.

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

Calibration is performed in accordance with written 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), after failing an operability check, 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. Instruments are calibrated with sources that are within 5% of the reference value and are traceable to the National Institute of Standards and Technology or equivalent.

~~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.~~

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4.7.1 Radiological Zones ~~Areas~~

~~Radiological zones Areas 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) control access to radioactive sources present in the facility. Table 4.1-2, Estimated Dose Rates, lists general dose rate estimates for the facility. These dose estimates were prepared based upon historical data from operating Urenco centrifuge enrichment facilities. Areas associated with higher dose rates may be restricted from general access, as determined by facility management. Areas where facility personnel spend substantial amounts of time are designed to minimize the exposure received (ALARA) when routine tasks are performed.~~

The following subsections describe how the facility Radiation Protection Program is implemented to protect site workers and the general public.

4.7.1.1 Unrestricted Area

NRC regulation 10 CFR 20.1003 (CFR, 2003n) 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 LES 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 (CFR, 2003o). 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 (CFR, 2003p), imposes

4.7 Radiation Surveys and Monitoring Programs Commitments

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. The Restricted Area boundary is the Controlled Access Area (CAA) security fence. This area has only one Entrance and Exit Control Point (EECP). All personnel are required to monitor themselves for contamination prior to exiting RCAs established within the Restricted Area.~~ ~~Restricted Areas that have the potential for contamination, using monitoring instruments that detect gross alpha contamination.~~

4.7.1.3 Radiologically Controlled Area (RCA)

An area within the Restricted Area where radiological hazards may exist that require progressive radiological access controls. 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. ~~Examples of Restricted Areas~~ areas within RCAs include storage areas for UF₆ and the potentially contaminated areas in the Cylinder Receipt and Dispatch Building. Personnel who have not been trained in radiation protection procedures are not allowed to access an RCA ~~Restricted Area~~ without escort by trained personnel.

The areas defined below may exist within an RCA ~~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 (CFR, 2003q).

- An area in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.05 mSv (5 mrem) in 1 hr at 30 cm (11.8 in) from the radiation source or from any surface that the radiation penetrates is designated a "Radiation Area" as defined in 10 CFR 20.1003 (CFR, 2003n).
- An "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 derived air concentrations (DACs) specified in Appendix B (CFR, 2003m), 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% 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" is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1 mSv (100 mrem) in 1 hour at 30 cm (11.8 in) 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 events).

4.7 Radiation Surveys and Monitoring Programs Commitments

- LES defines a "Contaminated Area" as an area where removable contamination levels are above 16.7 Bq/100 cm² (1,000 dpm/100 cm²) of alpha or beta/gamma activity.

The NRC limits the soluble uranium intake of an individual to 10 milligrams in a week in consideration of chemical toxicity. LES 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~~ 4.7.1.4 Controlled Area

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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 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 LES employees and who work only in the Controlled Area are subject to the exposure limits for members of the public (CFR, 2003b).

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~~ RCAs 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. Access control is by administrative methods and may be physically controlled for security reasons.

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~~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.~~

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Access to and egress from an RCA ~~Restricted Area~~ is through one of the monitor stations at the particular RCA ~~Restricted Area~~ boundary. Access to and egress from each Radiation Area, High Radiation Area, Contaminated Area or Airborne Radioactivity Area within an RCA ~~Restricted Area~~ may also be individually controlled. A monitor (frisker), step-off pad and container for any discarded protective clothing is provided as necessary at the egress point from these areas to prevent the spread of contamination.

Action levels for skin and personal clothing contamination at the point of egress from ~~Restricted Areas~~ an RCA and any additional designated areas within the ~~Restricted Area~~ an RCA (e.g., a Contaminated Area which is provided with a step-off pad and frisker) shall not exceed 16.7 Bq/100 cm² (1,000 dpm/100 cm²) of alpha or beta/gamma contamination. Clothing contaminated above egress limits shall not be released unless it can be decontaminated to within these limits. If skin or other parts of the body are contaminated above egress limits, reasonable steps shall be undertaken to effect decontamination.

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4.7.3 Posting for Radiation Protection Awareness

~~Restricted Areas~~RCAs and other areas within the ~~Restricted Areas~~RCAs (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 (CFR, 2003q). The radiation and contamination levels from the most recent survey are clearly noted on each posting.

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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 Contaminated Areas must wear appropriate protective clothing. If the areas containing the surface contamination can be isolated from adjacent work areas via a barrier such that dispersible material is not likely to be transferred beyond the area of contamination, personnel working in the adjacent area are not required to wear protective clothing. Areas requiring protective clothing are posted at each of their entry points.

Radiation protection management and associated technical staff are responsible for determining the need for protective clothing in each work area. Areas requiring protective clothing are identified by posting signs at all area entry points.

4.7.5 Personnel Monitoring for External Exposures

~~External exposures are received primarily from the radioactive decay products of ^{235}U and ^{238}U . Most notably these progeny are ^{231}Th (several gammas, all low energy and low abundance), ^{234}Th (several gammas, most low abundance and low energy), and ^{234}Pa and $^{234\text{m}}\text{Pa}$ (many gammas, variable abundance, low and high energy). The $^{234\text{m}}\text{Pa}$ is the primary gamma source and is expected to contribute to a significant portion of the external exposure. Over the life of the facility, the number of tails containing Uranium Byproduct Cylinders (UBCs) placed on the storage pad may increase to the pad's design capacity. In addition, the CRDB may reach its design capacity of feed and product cylinders. 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.~~

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If the individual is anticipated to receive a dose in excess of 10 CFR 20.1502 or it is required by the RWP, that individual will be issued a thermoluminescent dosimeters (TLD). All personnel whose duties routinely require them to enter ~~Restricted Areas~~an RCA wear individual external dosimetry devices, e.g., thermoluminescent dosimeters (TLDs) that are sensitive to beta, gamma and neutron radiation. Appropriate neutron survey meters are also available to the Radiation Protection staff. External dosimetry devices are evaluated at least quarterly~~an~~ established frequency (e.g. quarterly, semiannually, etc.) to ascertain external exposures. Administrative limits on radiation exposure are provided in Table 4.1-1, Administrative Radiation Exposure Limits.

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~~If 25% of the annual administrative limit (i.e., 2.5 mSv or 250 mrem) 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. The administrative limit already reflects~~

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~~ALARA principles, so this action level is appropriate. This investigation may include, but is not limited to procedural reviews, efficiency studies of the air handling system, cylinder storage protocol, and work practices.~~

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Anytime an administrative limit is exceeded, the Radiation Protection Manager is informed. The Radiation Protection Manager is responsible for determining the need for and recommending investigations or corrective actions to the responsible Manager(s). Copies of the 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 as required via direct bioassay (e.g. in vivo body counting), indirect bioassay (e.g., urinalysis), or an equivalent technique. For soluble (Class D) uranium, 10 CFR 20.1201(e) (CFR, 2003f) limits worker intake to no more than 10 milligrams of soluble uranium in a week. This is to protect workers from the toxic chemical effects of inhaling Class D uranium. The facility annual administrative limit for the Total Effective Dose Equivalent (TEDE) is 10 mSv (1000 mrem). Internal doses are evaluated at least annually.~~

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~~Continuous air monitoring in Airborne Radioactivity Areas is performed as necessary to complement the bioassay program. Alarm setpoints on the continuous air monitors in the Airborne Radioactivity Areas RCAs are 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. The action limit is based on ALARA principles. Other factors such as the biological elimination of uranium are considered. This investigation may include, but is not limited to procedural reviews, efficiency studies of the air handling system, and work practices.

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 (CFR, 2003r). Procedures for the evaluation and summation of doses are based on the guidance contained in Regulatory Guides 8.7 and 8.34.

4.7.8 Monitor Stations

~~Monitor stations are the entry and exit points for Restricted Areas RCAs. Monitors are provided to detect radioactive contamination on personnel and their personal items, including hard hats. All personnel are required to monitor themselves, any hand-carried personal items, and hard hats prior to exiting an RCA Restricted Area. Radiation protection management is responsible for Monitor Station provision and maintenance. Figure 4.7-2, Projected Radiological Zones shows the anticipated Restricted Areas. Monitor Station locations are evaluated and moved established as necessary in response to changes in the facility radiological conditions.~~

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4.7 Radiation Surveys and Monitoring Programs Commitments

4.7.9 Locker Rooms

Locker rooms for men and women are provided for personnel to change into appropriate work clothing and store personal belongings. The following facilities are provided for in the locker room area:

- Shower Rooms - shower rooms for men and women are provided as a place for personnel to wash/clean up after work. These shower rooms are not intended for personnel decontamination.
- Restrooms - restrooms for men and women are provided. These rooms are not for personnel decontamination.
- First Aid Station - a first aid station is provided to treat injured personnel.
- Personnel Decontamination Area - a personnel decontamination area is provided to handle cases of accidental radioactive contamination. A hand washing sink and a shower are provided for contamination removal.
- ~~Information Area - an information area is provided to notify personnel of information important to radiation protection.~~

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4.7.10 Storage Areas

Storage areas are provided for the following items:

- Protective (i.e., anti-contamination) clothing
- Respiratory protection equipment
- Shower rooms supplies
- Radiation protection supplies.

4.8 CONTAMINATION AND RADIATION CONTROL

The goal of maintaining occupational internal and external radiation exposures ALARA encompasses the individual's dose as well as the collective dose of the entire working population. Since the total effective dose equivalent (TEDE) is the sum of the internal and external exposures, the Radiation Protection Program addresses both contamination control and external radiation protection.

Listed below are examples of design and operating considerations that are implemented at the facility to reduce personnel radiation exposures:

- The enrichment process, with the exception of Liquid Sampling, is maintained under sub atmospheric pressure. The constant containment of UF₆ precludes direct contact with radioactive materials by personnel.
- Self-monitoring is required upon exit from ~~Restricted Areas~~ an RCA. Personnel are required to notify a member of the radiation protection staff if contamination is detected.
- All personnel are trained in emergency evacuation procedures in accordance with the facility Emergency Plan.
- 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.

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4.8.1 Internal Exposures

Because the radionuclides present in this facility under routine operations are primarily alpha and beta emitters (with some low-energy gamma rays), the potential for significant internal exposure is greater than that for external exposure. Parameters important to determining internal doses are:

- The quantity of radioactive material taken into the body
- The chemical form of the radioactive material
- The type and half-life of radionuclide involved
- The time interval over which the material remains in the body.

The principal modes by which radioactive material can be taken into the body are:

- Inhalation
- Ingestion
- Absorption through the skin
- Injection through wounds.

4.8 Contamination and Radiation Control

4.8.1.1 Bioassay

Internal radiological exposures are evaluated annually as noted in Section 4.7.7, Evaluation of Doses. Based on the results of air sample monitoring data, bioassays are performed for all personnel who are likely to have had an intake of one milligram of uranium during a week. This is 10% of the 10 mg (3.5 E-4 oz) in a week regulatory limit (10 CFR 20.1201(e) (CFR, 2003f)) for intake of Class D uranium. The bioassay program has a sensitivity of 5 µg/L (7 E-7 oz/gal) of uranium concentration, assuming that the sample is taken within ten days of the postulated intake and that at least 1.4 L (0.37 gal) of sample is available from a 24-hour sampling period. Until urinalysis results indicate less than 15 µg/L (2.0 E-6 oz/gal) of uranium concentration, workers are restricted from activities that could routinely or accidentally result in internal exposures to soluble uranium.

It might not be possible to achieve a sensitivity of 5 µg/L (7 E-7 oz/gal); if for example, all reasonable attempts to obtain a 1.4 L (0.37 gal) 24-hour sample within 10 days fail. In such a case, the sample is analyzed for uranium concentration (if measurable) and the worker's intake is estimated using other available data.

4.8.1.2 Air Monitoring and Sampling

Airborne activity in work areas is regularly determined in accordance with written procedures. Continuous air sampling in airborne radioactivity areas may be performed to complement the bioassay program. Using the values specified in 10 CFR 20 Appendix B (CFR, 2003m), if a worker could have inhaled radionuclide concentrations that are likely to exceed 12 DAC-hours in one week (seven days), then bioassay is conducted within 72 hours after the suspected or known exposure. Follow-up bioassay measurements are conducted to determine the committed effective dose equivalent. Until urinalysis results indicate less than 15 micrograms per liter uranium concentration, workers are restricted from activities that could routinely or accidentally result in internal exposures to soluble uranium.

Active on-line monitors for airborne alpha emitters are used to measure representative airborne concentrations of radionuclides that may be due to facility operation. On-line monitoring for gross alpha activity is performed assuming all the alpha activity is due to uranium. When airborne activity data is used for dose calculations, the assumption is that all the activity is due to ²³⁴U, class D material. The lower limit of detection is either 0.02 mg (7.16 E-7 oz) of uranium in the total sample or 3.7 nBq/mL (1 E-13 µCi/mL) gross alpha concentration. An action level is established at 1 mg (3.53 E-5 oz) of total uranium likely to be inhaled by a worker in seven days.

Monitors are permanently located in ~~Restricted Areas~~ RCAs. These permanent monitors are operated to collect continuous samples. When air sampling is conducted using continuous air sampling devices, the filters are changed and analyzed at the following frequencies:

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- Weekly and following any indication of release that might lead to airborne concentrations of uranium that are likely to exceed (1) 10% of the values listed in 10 CFR 20.1003 (CFR, 2003n), or (2) the total uranium action level of one milligram of total uranium inhaled in one week.
- Each Shift, following changes in process equipment or process control, and following detection of any event (e.g., leakage, spillage or blockage of process equipment) that are likely to exceed (1) 10% of the values listed in 10 CFR 20.1003 (CFR, 2003n),

4.8 Contamination and Radiation Control

- Decontamination
- Surveillance
- Procurement.

4.8.4 Instrumentation

~~Two~~ Three basic types of personnel monitoring equipment are used at the facility. These are count rate meters (as known as "friskers"), ~~and hand/foot monitors, and Personnel Contamination Monitors.~~

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~~4.8.5~~ 4.8.4.1 Friskers

Hand held friskers are typically placed in locations where conditions restrict the use of other monitors or for short-term use as necessary to ensure effective control of the spread of contamination. Instructions for the use of these instruments are posted in a prominent location near the instrument.

~~4.8.5~~ 4.8.4.2 Hand and Foot Monitors

These typically consist of multiple detectors arranged to monitor only hands and feet. Instructions for the use of these monitors are prominently posted on or near the instrument. Hand and foot monitors are used in applications where "pass-throughs" are frequent and where hand and foot monitoring is the major requirement. ~~Portal monitors, that can quickly scan large surface areas of the body, may be used where the number of personnel exiting an area, available space, etc., makes their use advantageous.~~

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4.8.4.3 Personnel Contamination Monitors (PCMs)

These typically consist of multiple detectors arranged to monitor the whole body. PCMs can quickly scan large surface areas of the body and may be used where the number of personnel existing an area, available space, etc., makes their use advantageous. The personnel monitor is placed at the control point for the RCA. Personnel existing the RCA are required to use the PCM for contamination on their body. If the PCM is out of service an alternative method of monitoring is required (e.g. friskers).

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~~4.8.6~~ 4.8.5 Contamination Control

Small contamination areas (~~i.e., less than one fourth of the room~~) may be roped off or otherwise segregated from the rest of an RCA Restricted Area. Appropriate clothing and/or other equipment is used to minimize exposure to radioactive material and prevent the spread of contamination. Provisions for monitoring contamination are discussed below. A contamination monitor (frisker), a step-off pad and a container for any discarded protective clothing may be placed at the access/egress point to the work area. The entire RCA Restricted Area is not posted as a Contaminated Area.

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~~4.8.6~~ 4.8.5.1 Surface Contamination

Contamination surveys are monitoring is performed infor all UF₆ process areas. Additional routine Ssurveys are performed include routine checks of non-UF₆ process areas, including areas normally not suspected to be contaminated. Monitoring includes direct radiation and

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4.8 Contamination and Radiation Control

removable contamination measurements. Survey procedures are based on the potential for contamination of an area and operational experience. ~~The Restricted Areas~~ Selected areas within RCA are surveyed at least weekly. The lunch room and change rooms are surveyed at least ~~daily~~ weekly.

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Removable surface contamination ~~is considered uranium contamination that is present on a surface and that can be transferred to a dry smear paper by rubbing with moderate pressure.~~ The facility uses various instruments such as proportional counters, alpha scintillation counters and thin window Geiger-Mueller tubes, to evaluate contamination levels.

If surface contamination levels exceed the following levels, clean-up of the contamination is initiated within 24 hours of the completion of the analysis:

- Removable contamination: 83.3 Bq/100 cm² (5000 dpm/100 cm²) alpha or beta/gamma
- Fixed contamination: 4.2 kBq/100 cm² (250,000 dpm/100 cm²) alpha or beta/gamma

4.9 MAINTENANCE AREAS-METHODS AND PROCEDURES FOR CONTAMINATION CONTROL

Designing processes and equipment that contain radioactive material to require as little maintenance as possible ensures that personnel radiation exposures are ALARA. Additional exposure reductions are achieved by:

- A. Removing as much radioactive material as possible from the equipment and the area prior to maintenance, thereby reducing the intensity of the radiation field
- B. Providing adequate space for ease of maintenance reducing the length of time required to complete the task, thereby reducing the time of exposure
- C. Preparing and using procedures that contain specifications for tools and equipment needed to complete the job
- D. Proper job planning, including practice on mockups
- E. Previews of previous similar jobs
- F. Identification and communication of the highest contamination areas to the workers prior to the start of work.

4.9.1 Decontamination Workshop

(See ~~12.1.3 C~~ and ~~12.3 A~~ 12.1.1.3.3 and 12.1.3.4) The Decontamination Workshop and Decontamination System are located in the same room in the CRDB. This room is called the Decontamination Workshop. The Decontamination Workshop contains an area to break down and strip contaminated equipment and to decontaminate the equipment and its components. The decontamination systems in the workshop are designed to remove radioactive contamination from contaminated materials and equipment. The only significant forms of radioactive contamination found in the facility are uranium hexafluoride (UF₆), uranium tetrafluoride (UF₄) and uranyl fluoride (UO₂F₂).

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One of the functions of the Decontamination Workshop is to provide a maintenance facility for both UF₆ pumps and for vacuum pumps. The workshop is used for the temporary storage and subsequent dismantling of failed pumps. The dismantling area is in physical proximity to the decontamination train, in which the dismantled pump components are processed.

The process carried out within the Decontamination Workshop begins with receipt and storage of contaminated pumps, out-gassing, Perfluorinated Polyether (PFPE) oil removal and storage, and pump stripping. The dismantling, maintenance, and decontamination of other plant components besides pumps is also routine and includes valves, piping, instruments, sample bottles, tools, and scrap metal. Personnel entry into the facility is via a sub-change facility. This area has the required contamination area access controls, washing and monitoring facilities.

The decontamination part of the process consists of a series of steps following equipment disassembly including degreasing, decontamination, drying, and inspection. Items from uranium hexafluoride systems, waste handling systems, and miscellaneous other items are decontaminated in this system.

4.9 Maintenance Areas-Methods and Procedures for Contamination Control

4.9.2 Contaminated Material Handling Room

The Contaminated Material Handling Room, located in the CRDB, provides an area for the Recycling Group to store protective clothing drums and other material/waste containers that have been assayed and released from the Safeguards item control program. This area will normally provide storage for containers awaiting Radiation Protection survey to be either unconditionally released or transferred to the solid waste collection system for additional processing. In addition, the Contaminated Material Handling Room will contain cabinets and bins with supplies to support the waste program and a connection to the CRDB GEVS to support ventilation engineering controls when required.

4.9.3 Personnel Contamination Monitor (ARGOS)

The ARGOS personnel monitor is placed at the control point for the RCA. Personnel exiting the RCA are required to use the ARGOS to monitor for contamination on their body. If the ARGOS is out of service and another ARGOS is unavailable, an alternative method of monitoring is required (e.g. friskers)

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4.12 References

CFR, 2003q. Title 10, Code of Federal Regulation, Section 20.1902, Posting requirements, 2003.

CFR, 2003r. Title 10, Code of Federal Regulations, Section 20.1202, Compliance with requirements for summation of external and internal does, 2003.

CFR, 2003s. Title 10, Code of Federal Regulations, Section 70.74, Additional reporting requirements, 2003.

CFR, 2003t. Title 10, Code of Federal Regulations, Section 20.2202, Method for obtaining approval of proposed disposal procedures, 2003.

CFR, 2003u. Title 10, Code of Federal Regulations, Section 20.2206, Transfer for disposal and manifests, 2003.

CFR 2003v. Title 10, Code of Federal Regulations, Section 20.2103, Records of Surveys, 2003.

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Urenco, 2000. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2000.

Urenco, 2001. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2001.

Urenco, 2002. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2002.

4.13 CHAPTER 4 TABLES

Table 4.1-1 Administrative Radiation Exposure Limits

	Administrative Limit
Total Effective Dose Equivalent (TEDE)	10 mSv/yr (1000 mrem/yr)

Notes:

A.a) Excludes accident situations

B.b) No routine extremity or skin monitoring is required

C.c) TEDE is the sum of internal dose and external dose received during routine operations

D.d) NRC limit is 50 mSv/yr (5000 mrem/yr)

Table 4.1-2 Estimated Dose Rates

Area or Component	Dose Rate, mSv/hr (mrem/hr)
Plant general area (excluding Separations Building Module)	< 1 E-4 (< 0.01)
Separations Building Module – Cascade Halls	5 E-4 (0.05)
Separations Building Module –UF ₆ Handling Area & Process Services Corridor	1 E-3 (0.1)
Empty used UF ₆ shipping cylinder	0.1 on contact (10.0) 0.01 at 1 m (1.0)
Full UF ₆ shipping cylinder	0.05 on contact (5.0) 2 E-3 at 1 m (0.2)

Table 4.1-3 Estimated Individual Exposures

Position	Annual Dose ^(a) mSv (mrem)
General Office Staff	< 0.05 (< 5.0)
Typical Operations & Maintenance Technician	1 (100)
Typical Cylinder Handler	3 (300)

(a) The average worker exposure at the Urenco Capenhurst facility during the years 1998 through 2002 was approximately 0.2 mSv (20 mrem) (Urenco, 2000; Urenco, 2001; Urenco, 2002)

4.13 Chapter 4 Tables

Table 4.11-1 Material Quantities		
Source and/or Special Nuclear Material	Physical Form	Maximum Amount to be Processed at Any One Time (μCi)
Cl-36	Unsealed, any form	2.26E-1
Cr-51	Sealed per §30.32(g)(1)	1.00E+1
Co-57	Sealed per §30.32(g)(1)	1.00E+4
Co-60	Sealed per §30.32(g)(1)	1.00E+1
Ni-63	Unsealed, any form	1.00E+1
Sr-85	Sealed per §30.32(g)(1)	1.00E+1
Y-88	Sealed per §30.32(g)(1)	1.00E+1
Sr-90	Unsealed, any form Sealed per §30.32(g)(1)	5.00E+0
Y-90	Unsealed, any form	5.00E+0
Tc-99	Unsealed, any form	1.00E+1
Cd-109	Sealed per §30.32(g)(1)	1.00E+3
Sn-113	Sealed per §30.32(g)(1)	1.00E+1
Tc-123m	Sealed per §30.32(g)(1)	1.00E+1
Cs-137	Sealed per §30.32(g)(1)	5.00E+4
Eu-152 (13y)	Sealed per §30.32(g)(1)	2.00E+0
Po-210	Unsealed, any form Sealed per §30.32(g)(1)	1.00E+1
Th-230	Sealed per §30.32(g)(1) Unsealed, any form	1.00E+10
U-232	Sealed per §30.32(g)(1) Unsealed, any form	1.00E+10
U-233	Sealed per §30.32(g)(1)	1.00E+15
U-234	Sealed per §30.32(g)(1) Unsealed, any form	1.00E+10
U-235	Sealed per §30.32(g)(1) Unsealed, any form	1.00E+10
U-236	Sealed per §30.32(g)(1)	1.00E+15
U-238	Sealed per §30.32(g)(1) Unsealed, any form	1.00E+10
Am-241	Sealed per §30.32(g)(1)	5.00E+4
Cf-252	Sealed per §30.32(g)(1)	5.00E+24

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4.13 Chapter 4 Tables

Table 4.13-1 Material Quantities		
Source and/or Special Nuclear Material	Physical Form	Maximum Amount to be Possessed at Any One Time (μCi)
Ce-139	Sealed per §30.32(g)(1)	1.00E+1

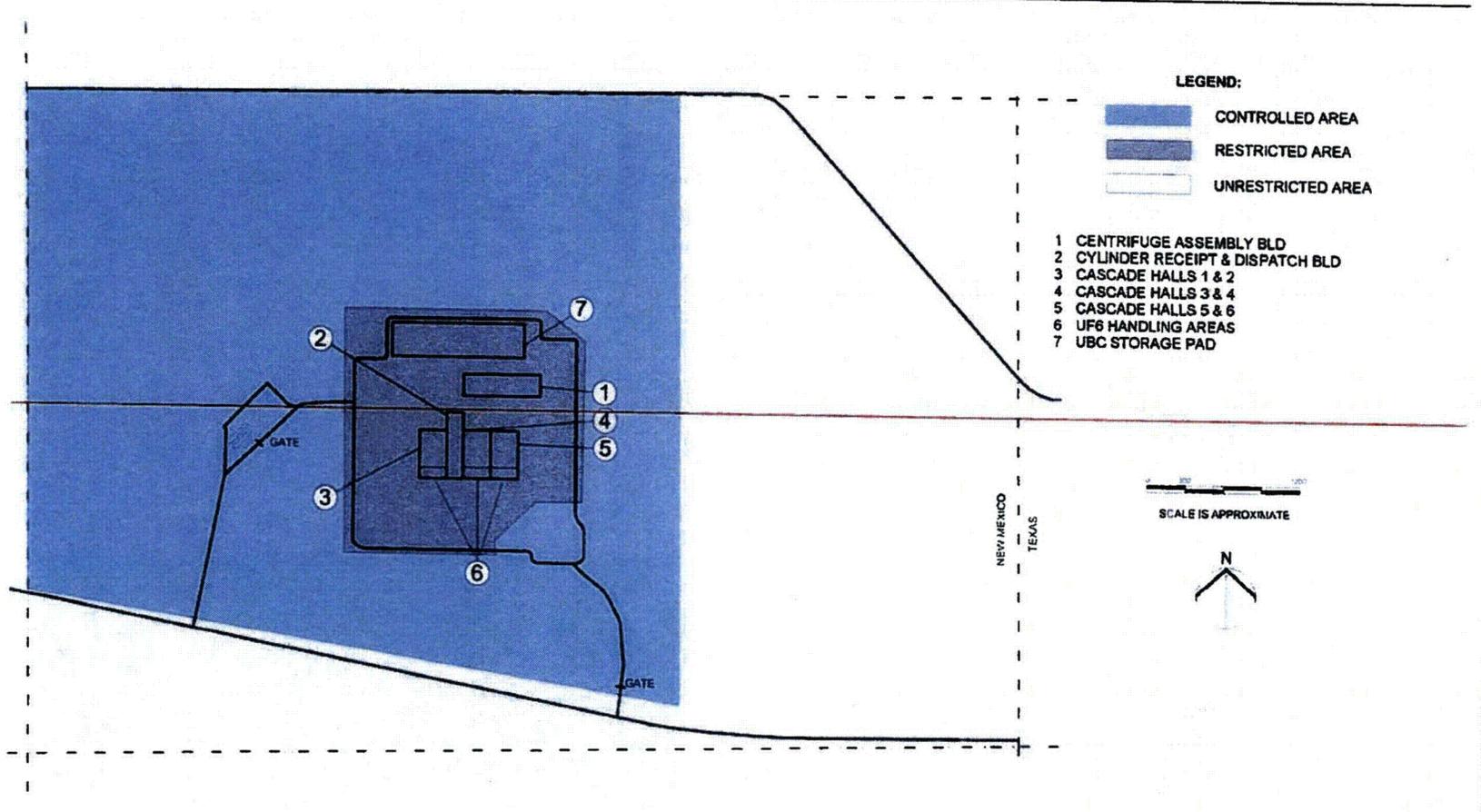
For limits of possession for radioactive material types, quantities, and forms see current version of SNM-2010.

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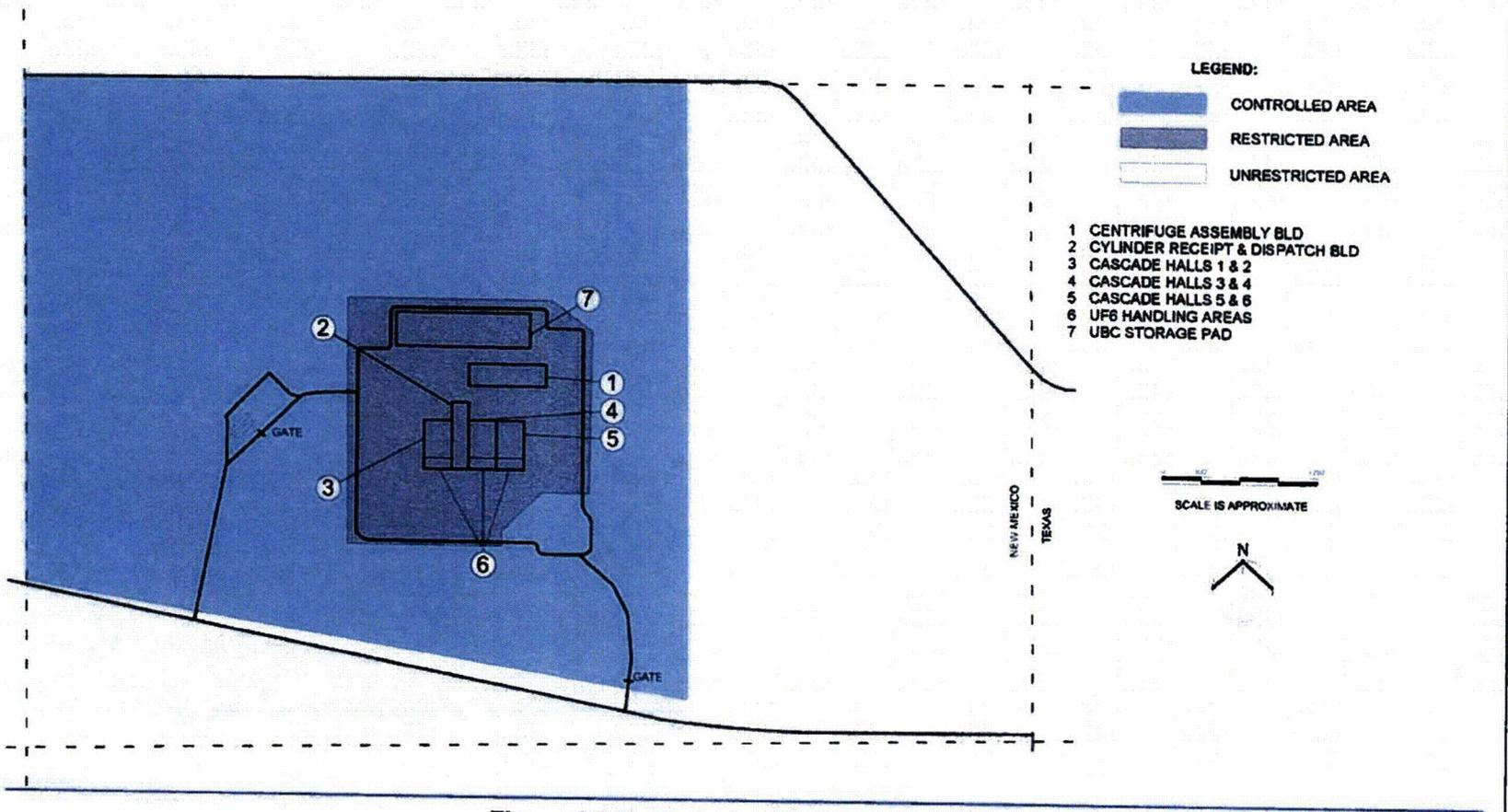
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4.14 Chapter 4 Figures



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4.14 Chapter 4 Figures



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Figure 4.7-2 Projected Radiological Zones

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5.1 THE NUCLEAR CRITICALITY SAFETY (NCS) PROGRAM

The facility has been designed and will be constructed and operated such that a nuclear criticality event is prevented, and to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a). Nuclear criticality safety at the facility is assured by designing the facility, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and any credible accident. Items Relied On For Safety (IROFS) identified to ensure subcriticality are discussed in the NEF Integrated Safety Analysis Summary.

5.1.1 Management of the Nuclear Criticality Safety (NCS) Program

The NCS criteria in Section 5.2, Methodologies and Technical Practices, are used for managing criticality safety and include adherence to the double contingency principle as stated in the ANSI/ANS-8.1, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors. 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 NEF meets the double contingency principle. The NEF meets the double contingency principle in that process design incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

The plant will produce no greater than 5.0 % enrichment. However, as additional conservatism, ~~the most nuclear criticality safety analyses for enriched material are performed assuming a ²³⁵U enrichment of 6.0 %~~, ~~except for Contingency Dump System traps which are analyzed assuming a ²³⁵U enrichment of 1.5 %~~, and include appropriate margins to safety. The exceptions to this are the systems and components associated with a cascade dump which are analyzed assuming 1.5 %. These include the Contingency Dump System equipment and piping on the 2nd floor of the Process Services Area and the Tails Take-off System. In accordance with 10 CFR 70.61(d) (CFR, 2003b), the general criticality safety philosophy is to prevent accidental uranium enrichment excesses, provide geometrical safety when practical, provide for moderation controls within the UF₆ processes and impose strict mass limits on containers of aqueous, solvent based, or acid solutions containing uranium. Interaction controls provide for safe movement and storage of components. Plant and equipment features assure prevention of excessive enrichment. The plant is divided into distinctly separate Assay Units (called Cascade Halls) with no common UF₆ piping. UF₆ blending is done in a physically separate portion of the plant. Process piping, individual centrifuges and chemical traps other than the contingency dump chemical traps, are safe by limits placed on their diameters. Product cylinders rely upon uranium enrichment, moderation control and mass limits to protect against the possibility of a criticality event. Each of the liquid effluent collection tanks that hold uranium in solution is mass controlled, as none are geometrically safe. As required by 10 CFR 70.64(a) (CFR, 2003c), by observing the double contingency principle throughout the plant, a criticality accident is prevented. In addition to the double contingency principle, effective management of the NCS Program includes:

- An NCS program to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a) will be developed, implemented, and maintained.
- Safety parameters and procedures will be established.
- The NCS program structure, including definition of the responsibilities and authorities of key program personnel will be provided.

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5.1 The Nuclear Criticality Safety (NCS) Program

assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2 as described in Section 5.2.1.3.3, Uranium Accumulation and Moderation Assumption. A uniform aqueous solution of UO_2F_2 , and a fixed enrichment are conservatively modeled using MONK8A (SA, 2001) and the JEF2.2 library. Criticality analyses were performed to determine the maximum value of a parameter to yield $k_{eff} = 1$. The criticality analyses were then repeated to determine the maximum value of the parameter to yield a $k_{eff} = 0.95$. Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2 , shows both the critical and safe limits for 5.0 % and 6.0 %.

Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, lists the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2 , which are used as control parameters to prevent a nuclear criticality event. Although the NEF will be limited to 5.0 % enrichment, as additional conservatism, the values in Table 5.1-2, Safety Criteria for Buildings/Systems/ Components, represent the limits based on 6.0 % enrichment except for the Contingency Dump System ~~traps equipment and piping on the 2nd floor of the Process Services Area and the Tails Take-off System~~ which are limited to 1.5 % ^{235}U .

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The values on Table 5.1-1 are chosen to be critically safe when optimum light water moderation exists and reflection is considered within isolated systems. The conservative modeling techniques provide for more conservative values than provided in ANSI/ANS-8.1. The product cylinders are only safe under conditions of limited moderation and enrichment. In such cases, both design and operating procedures are used to assure that these limits are not exceeded.

All Separation Plant components, which handle enriched UF_6 , other than the Type 30B cylinders and the first stage UF_6 pumps and contingency dump chemical traps, are safe by geometry. Centrifuge array criticality is precluded by a probability argument with multiple operational procedure barriers. Total moderator or H/U ratio control as appropriate precludes product cylinder criticality.

In the Cylinder Receipt and Dispatch Building criticality safety for uranium loaded liquids is ensured by limiting the mass of uranium in any single tank to less than or equal to 12.2 kg U (26.9 lb U). Individual liquid storage bottles are safe by volume. Interaction in storage arrays is accounted for.

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

Enrichment

~~Enrichment is controlled to limit the percent ^{235}U within any process, vessel, or container, except the contingency dump system, to a maximum enrichment of 5 %.~~ The design of the contingency dump system controls enrichment to a limit of 1.5 % ^{235}U . ~~Although NEF is limited to a maximum enrichment of 5 % as added conservatism nuclear criticality safety is analyzed using an enrichment of 6 % ^{235}U .~~ Enrichment is controlled to limit the percent ^{235}U within any process vessel or container to a maximum of 5% except for the systems and components associated with a cascade dump. For added conservatism the systems controlled to 5% are analyzed at 6%.

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Assuming a product enrichment of 6% limits the upper bound for the average cascade enrichment to less than 1.5%, the systems and components associated with a cascade dump (Tails Take-off System, Contingency Sump System) are conservatively analyzed at 1.5%

5.1 The Nuclear Criticality Safety (NCS) Program

Geometry/Volume

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

The geometry/volume limits are chosen to ensure $k_{\text{eff}} = k_{\text{calc}} + 3 \sigma_{\text{calc}} < 0.95$.

The safe values of geometry/volume in Table 5.1-1 define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming 6 % ^{235}U for safety margin.

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Moderation

Water and oil are the moderators considered in NEF. At NEF the only system where moderation is used as a control parameter is in the product cylinders. Moderation control is established consistent with the guidelines of ANSI/ANS-8.22 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 assure 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 a 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.

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. Analysis or sampling is employed to verify the mass of the material. Conservative administrative limits for each operation are specified in the operating procedures.

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.

5.1 The Nuclear Criticality Safety (NCS) Program

atmospheres. It is highly unlikely that any size changes of process piping, cylinders, cold traps, or chemical traps under these conditions, would lead to a criticality situation because a volume or mass limit may be exceeded.

Within the Separations Building, significant accumulations of enriched UF₆ reside only in the Product Low Temperature Take-off Stations, Product Liquid Sampling Autoclaves, Product Blending System or the UF₆ cold traps. All these, except the UF₆ cold traps, contain the UF₆ in 30B cylinders. All these significant accumulations are within enclosures protecting them from water ingress. The facility design has minimized the possibility of accidental moderation by eliminating direct water contact with these cylinders of accumulated UF₆. In addition, the facility's stringent procedural controls for enriching the UF₆ assure that it does not become unacceptably hydrogen moderated while in process. The plant's UF₆ systems operating procedures contain safeguards against loss of moderation control (ANSI/ANS 8.22). No neutron poisons are relied upon to assure criticality safety.

5.1.4 Description of Safety Criteria

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO₂F₂, are applied to the facility to prevent a nuclear criticality event. Although the NEF will be limited to 5.0 w/o enrichment, as additional conservatism, the values in Table 5.1-2, represent the limits based on 6.0 w/o enrichment with the exception of the Tails Take-off and Contingency Dump Systems. These systems are limited to the maximum process system average enrichment, 1.5%.

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Where there are significant in-process accumulations of enriched uranium as UF₆, the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

5.1.5 Organization and Administration

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

The Criticality Safety Officer reports to the Health and Safety Manager as described in Chapter 2, Organization and Administration. The Health and Safety Manager is accountable for overall criticality safety of the facility, is administratively independent of production responsibilities, and has the authority to shut down potentially unsafe operations.

Designated responsibilities of the Criticality Safety Officer include the following:

- Establish the Nuclear Criticality Safety Program, including design criteria, procedures, and training
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters, with input from the Criticality Safety Engineers

5.1 The Nuclear Criticality Safety (NCS) Program

- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs) (i.e., non-calculation engineering judgments regarding whether existing criticality safety analyses bound the issue being evaluated or whether new or revised safety analyses are required)
- Specify criticality safety control requirements and functionality
- Provide advice and counsel on criticality safety control measures
- Support emergency response planning and events
- Evaluate the effectiveness of the Nuclear Criticality Safety Program using audits and assessments
- Provide criticality safety postings that identify administrative controls for operators in applicable work areas.

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Criticality Safety Engineers will be provided in sufficient number to support the program technically. They are responsible for the following:

- Provide criticality safety support for integrated safety analyses and configuration control
- Perform NCS analyses (i.e., calculations), write NCS evaluations, and approve proposed changes in process conditions on equipment involving fissionable material

Qualified Criticality Safety Engineers may also perform tasks associated with Criticality Safety program implementation and assessment.

The minimum qualifications for the Criticality Safety Officer and the Criticality Safety Engineer are described in Section 2.2.4. The Health and Safety Manager has the authority and responsibility to assign and direct activities for the Criticality Safety Program. The Criticality Safety Officer is responsible for implementation of the NCS program.

The NEF implements the intent of the administrative practices for criticality safety, as contained in Section 4.1.1 of American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors. A policy will be established whereby personnel shall report defective NCS conditions and perform actions only in accordance with written, approved procedures. Unless a specific procedure deals with the situation, personnel shall report defective NCS conditions and take no action until the situation has been evaluated and recovery procedures provided.

5.2 Methodologies and Technical Practices

In accordance with the guidance in NUREG-1520, code validation for the specific application has been performed (see AREVA in ISAS table 3.0-1). Specifically, the experiments provided in Table 5.2-1, Uranium Experiments Used for Validation, were calculated and documented in the MONK8A Validation and Verification report (see AREVA in ISAS table 3.0-1) for the National Enrichment Facility. In addition, the MONK8A Validation and Verification report (see AREVA in ISAS table 3.0-1) satisfies the commitment to ANSI/ANS-8.1 and includes details of computer codes used, operations, recipes for choosing code options (where applicable), cross sections sets, and any numerical parameters necessary to describe the input.

The MONK8A computer code and JEF2.2 library are within the scope of the Quality Assurance Program.

5.2.1.2 Limits on Control and Controlled Parameters

The validation process established a bias by comparing calculations to measured critical experiments. With the bias determined, an upper safety limit (USL) can be determined using the following equation from NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology:

$$USL = 1.0 + \text{Bias} - \sigma_{\text{Bias}} - \Delta_{\text{SM}} - \Delta_{\text{AOA}}$$

Where the critical experiments are assumed to have a k_{eff} of unity, and the bias was determined by comparison of calculation to experiment. From Section 5.2.1.1, Methods Validation, the bias is positive and since a positive bias may be non-conservative, the bias is set to zero. The σ_{Bias} from the MONK8A Validation and Verification (see AREVA in ISAS table 3.0-1) is 0.0085 and a value of 0.05 is assigned to the subcritical margin, Δ_{SM} . The term Δ_{AOA} is an additional subcritical margin to account for extensions in the area of applicability. Since the experiments in the benchmark are representative of the application, the term Δ_{AOA} is set to zero for systems and components not associated with the Contingency Dump System. For the Contingency Dump System, it was necessary to extrapolate the area of applicability to include 1.5% enrichment and the term Δ_{AOA} is set to 0.0014 to account for this extrapolation. Thus, the USL becomes:

- $USL = 1 + 0 - 0.0085 - 0.05 = 0.9415$ (for systems and components NOT associated with the Contingency Dump System)
- $USL = 1 + 0 - 0.0085 - 0.05 - 0.0014 = 0.9401$ (for the Contingency Dump System and Tails Take-off System)

NUREG/CR-6698 indicates that the following condition be demonstrated for all normal and credible abnormal operating conditions:

$$k_{\text{calc}} + 2 \sigma_{\text{calc}} < USL$$

The risk of an accidental criticality resulting from NEF operations is inherently low. The low risk warrants the use of an alternate approach.

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5.2 Methodologies and Technical Practices

At the low enrichment limits established for the NEF, sufficient mass of enriched uranic material cannot be accumulated to achieve criticality without moderation. Uranium in the centrifuge plant is inherently a very dry, unmoderated material. Centrifuge separation operations at NEF do not include solutions of enriched uranium. For most components that form part of the centrifuge plant or are connected to it, sufficient mass of moderated uranium can only accumulate by reaction between UF_6 and moisture in air leaking into plant process systems, leading to the accumulation of uranic breakdown material. Due to the high vacuum requirements for the normal operation of the facility, air inleakage into the process systems is controlled to very low levels and thus the highly moderated condition assumed represents an abnormal condition. In addition, excessive air in-leakage would result in a loss of vacuum, which in turn would cause the affected centrifuges to crash (self destruct) and the enrichment process in the affected centrifuges to stop. As such, buildup of additional mass of moderated uranic breakdown material, such that component becomes filled with sufficient mass of enriched uranic material for criticality, is precluded. Even when accumulated in large UF_6 cylinders or cold traps, neither UF_6 nor UO_2F_2 can achieve criticality without moderation at the low enrichment limit established for the NEF.

Therefore, due to the low risk of accidental criticality associated with NEF operations and the margin that exists in the design and operation of the NEF with respect to nuclear criticality safety, a margin of subcriticality for safety of 0.05 (i.e., $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$) is adequate to ensure subcriticality is maintained under normal and abnormal credible conditions. As such, the NEF will be designed using the equation:

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

5.2.1.3 General Nuclear Criticality Safety Methodology

The NCS analyses results provide values of k-effective (k_{eff}) to conservatively meet the upper safety limit. The following sections provide a description of the major assumptions used in the NCS analyses.

5.2.1.3.1 Reflection Assumption

The layout of the NEF is a very open design and it is not considered credible that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. Full water reflection of vessels has therefore been discounted. However, where appropriate, spurious reflection due to walls, fixtures, personnel, etc. has been accounted for by assuming 2.5 cm (0.984 in) of water reflection around vessels.

5.2.1.3.2 Enrichment Assumption

~~The NEF will operate with a 5.0 % ^{235}U enrichment limit. However, the nuclear criticality safety calculations used an enrichment of 6.0 % ^{235}U . This assumption provides additional conservatism for plant design. Enrichment is controlled to limit the percent ^{235}U within any process vessel or container to a maximum of 5% except for the systems and components associated with a cascade dump. For added conservatism the systems controlled to 5% are analyzed at 6%.~~

Assuming a product enrichment is 6% limits the upper bound for the average cascade enrichment to less than 1.5% the systems and components associates with a cascade dump (Tails Take-off System, Contingency Dump System) are conservatively analyzed at 1.5%

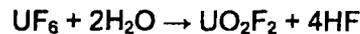
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5.2 Methodologies and Technical Practices

5.2.1.3.3 Uranium Accumulation and Moderation Assumption

Most components that form part of the centrifuge plant or are connected to it assume 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 nuclear criticality safety analyses documentation). The ratio is based on the assumption that significant quantities of moderated uranium could only accumulate by reaction between UF₆ and moisture in air leaking into the plant. 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. The maximum H/U ratio of 7 for the uranyl fluoride-water mixture is derived as follows:

The stoichiometric reaction between UF₆ and water vapor in the presence of excess UF₆ can be represented by the equation:



Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound. Experimental studies (Lychev, 1990) suggest that solid hydrates of compositions UO₂F₂·1.5H₂O and UO₂F₂·2H₂O can form in the presence of water vapor, the former composition being the stable form on exposure to atmosphere.

It is assumed that the hydrate UO₂F₂·1.5H₂O is formed and, additionally, that the HF produced by the UF₆/water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:



For the MONK8A (SA, 2001) calculations, the composition of the breakdown product was simplified to UO₂F₂·3.5H₂O that gives the same H/U ratio of 7 as above.

In the case of oils, UF₆ pumps and vacuum pumps use a fully fluorinated perfluorinated polyether (PFPE) type lubricant. Mixtures of UF₆ and PFPE oil would be a less conservative case than a uranyl fluoride/water mixture, since the maximum HF solubility in PFPE is only about 0.1 %/o. Therefore, the uranyl fluoride/water mixture assumption provides additional conservatism in this case.

5.2.1.3.4 Vessel Movement Assumption

~~The interaction controls placed on movement of vessels containing enriched uranium are specified in the facility procedures. In general, any item in movement (an item being either an individual vessel or a specified batch of vessels) must be maintained at 60 cm (23.6 in) edge separation from any other enriched uranium, and that only one item of each type, e.g., one trap and one pump, may be in movement at one time. These spacing restrictions are relaxed for vessels being removed from fixed positions. In this situation, one vessel may approach an adjacent fixed plant vessel/component without spacing restrictions. The limits placed on movement of an individual vessel or a specified batch of vessels containing enriched uranium are specified in the facility procedures or work plans, both of which are reviewed by Nuclear Criticality Safety. Specified limits may not be required based on bounding or process/system-specific NCS evaluations or analysis.~~

Of the subset of individual vessels or groups of vessels that do not have specified controls but are bounded by a the single-parameter SBD limits in Table 5.1-1, separation must be maintained at least 60 cm (23.6 in) from any other enriched uranium.

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5.2 Methodologies and Technical Practices

Vessels or groups of vessels that do not comply with either of the statements above must not be moved without the written approval of the Criticality Safety Officer.

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5.2.1.3.5 Pump Free Volume Assumption

There are two types of pumps used in product and dump systems of the plant:

- The vacuum pumps (product and dump) are rotary vane pumps. In the enrichment plant fixed equipment, these are assumed to have a free volume of 14 L (3.7 gal) and are modeled as a cylinder in MONK8A (SA, 2001). This adequately covers all models likely to be purchased.
- The UF₆ pumping units are a combination unit of two pumps, one 500 m³/hr (17,656 ft³/hr) pump with a free volume of 8.52 L (2.25 gal) modeled as a cylinder, and a larger 2000 m³/hr (70,626 ft³/hr) pump which is modeled explicitly according to manufacturer's drawings.

5.2.1.4 Nuclear Criticality Safety Analyses

Nuclear criticality safety is analyzed for the design features of the plant system or component and for the operating practices that relate to maintaining criticality safety. The analysis of individual systems or components and their interaction with other systems or components containing enriched uranium is performed to assure the criticality safety criteria are met. The nuclear criticality safety analyses and the safe values in Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO₂F₂, provide a basis for the plant design and criticality hazards identification performed as part of the Integrated Safety Analysis.

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safe values of Table 5.1-1, are applied to the facility design to prevent a nuclear criticality event. The NEF is designed and operated in accordance with the parameters provided in Table 5.1-2. The Integrated Safety Analysis reviewed the facility design and operation and identified Items Relied On For Safety to ensure that criticality does not pose an unacceptable risk.

Where there are significant in-process accumulations of enriched uranium as UF₆ the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

Each NCS analysis includes, as a minimum, the following information.

- A discussion of the scope of the analysis and a description of the system(s)/process(es) being analyzed.
- A discussion of the methodology used in the criticality calculations, which includes the validated computer codes and cross section library used and the k_{eff} limit used (0.95).
- A discussion of assumptions (e.g. reflection, enrichment, uranium accumulation, moderation, movement of vessels, component dimensions) and the details concerning the assumptions applicable to the analysis.
- A discussion on the system(s)/process(es) analyzed and the analysis performed, including a description of the accident or abnormal conditions assumed.

5.6 CHAPTER 5 TABLES

Table 5.1-1 Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2

Parameter	Critical Value $k_{eff} = 1.0$	Safe Value $k_{eff} = 0.95$	Safety Factor
Values for 5.0 % enrichment			
Volume	28.9 <u>30.3</u> L (7.6 <u>8.0</u> gal)	24.6 <u>22.9</u> L (5.7 <u>6.1</u> gal)	0.75 <u>0.76</u>
Cylinder Diameter	26.2- <u>26.6</u> cm(10.5 <u>3</u> in)	23.6 <u>23.9</u> cm (9.4 <u>3</u> in)	0.90
Slab Thickness	42.6 <u>12.8</u> cm (5.0 in)	40.7 <u>11.1</u> cm (4.2 in)	0.85 <u>0.87</u>
Water Mass	47.3 <u>18.5</u> kg H ₂ O (38.4 <u>40.8</u> lb H ₂ O)	42.7 <u>14.2</u> kg H ₂ O (28.0 <u>31.1</u> lb H ₂ O)	0.73 <u>0.77</u>
Areal Density	44.9- <u>11.8</u> g/cm ² (24.4- <u>24.2</u> lb/ft ²)	9.8- <u>9.9</u> g/cm ² (20.1- <u>20.3</u> lb/ft ²)	0.82 <u>0.84</u>
Uranium Mass	37- <u>36.7</u> kg U (84.6 <u>80.9</u> lb U)		
- no double batching		26.6 <u>26.8</u> kg U (58.6 <u>59.1</u> lb U)	0.72 <u>0.73</u>
- double batching		46.6 <u>16.5</u> kg U (36.6 <u>36.4</u> lb U)	0.45
Values for 6.0 % enrichment			
Volume	24.4- <u>25.3</u> L (6.3- <u>6.7</u> gal)	48- <u>19.3</u> L (4.8- <u>5.1</u> gal)	0.75 <u>0.76</u>
Cylinder Diameter	24.4- <u>24.8</u> cm (9.6- <u>9.8</u> in)	21.9- <u>22.4</u> cm (8.6- <u>8.8</u> in)	0.90
Slab Thickness	41.5- <u>11.6</u> cm (4.5- <u>4.6</u> in)	9.9- <u>10.1</u> cm (3.9- <u>4.0</u> in)	0.86 <u>0.87</u>
Water Mass	15.4 kg H ₂ O (34.0 lb H ₂ O)	44.5 <u>11.9</u> kg H ₂ O (25.4 <u>26.2</u> lb H ₂ O)	0.75 <u>0.77</u>
Areal Density	9.5 <u>9.4</u> g/cm ² (49.5 <u>19.3</u> lb/ft ²)	7.5 <u>7.9</u> g/cm ² (15.4- <u>16.2</u> lb/ft ²)	0.79 <u>0.84</u>
Uranium Mass	27 kg U (59.5 lb U)		
- no double batching		20.1 kg U (29.7 kg UF ₆)	0.72 <u>0.74</u>
- double batching		12.2 kg U (26.9 lb U)	0.45

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Table 5.1-2 Safety Criteria for Buildings/Systems/Components

Building/System/Component	Control Mechanism	Safety Criteria
Enrichment	Enrichment	5.0 w/o (6 w/o ²³⁵ U used in NCS)
Centrifuges	Diameter	< 24.922.4 cm (8.68.8 in)
Product Cylinders (30B)	Moderation	H < 0.950.98 kg (2.092.16 lb)
UF ₆ Piping	Diameter	< 24.922.4 cm (8.68.8 in)
Chemical Traps	Diameter	< 24.922.4 cm (8.68.8 in)
Product Cold Trap	Diameter	< 24.922.4 cm (8.68.8 in)
Contingency Dump System Traps/Tails System	Enrichment	1.5 w/o ²³⁵ U (used in NCS)
Tanks	Mass	< 12.2 kg U (26.9 lb U)
Feed Cylinders	Enrichment	< 0.72 w/o ²³⁵ U
Uranium Byproduct Cylinders	Enrichment	< 0.72 w/o ²³⁵ U
UF ₆ Pumps (first stage)	N/A	Safe by explicit calculation
UF ₆ Pumps (second stage)	Volume	< 48.019.3 L (4.85.1 gal)
Individual Uranic Liquid Containers, e.g., PFPE Oil Bottle, Laboratory Flask, Mop Bucket	Volume	< 48.019.3 L (4.85.1 gal)
Vacuum Cleaners Oil Containers	Volume	< 48.019.3 L (4.85.1 gal)

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6.1 CHEMICAL INFORMATION

This section addresses the criteria utilized to classify all site chemicals based on their potential for harm and as defined by regulatory requirements. It also presents information on the properties of ~~these~~ selected chemicals. Chemical formulas in this Chapter utilize subscripting per standard convention.

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6.1.1 Chemical Screening and Classification

~~Table 6.1-1, Chemicals—Hazardous Properties, provides the listing of chemicals and related chemical wastes that are expected to be in use at the NEF. Chemical formulas in this Chapter utilize subscripting per standard convention. The hazardous properties of each chemical and related chemical waste have been listed. Also, each chemical or related waste has been classified into one of three categories (NEF Classes): Chemicals of Concern (Class 1), Interaction Chemicals (Class 2), or Incidental Chemicals (Class 3). A Chemical Safety Program tracks the general locations of hazardous chemicals onsite and the specific hazards associated with each chemical. Each chemical at the NEF has been classified into one of three categories (NEF Classes): Chemicals of Concern (Class 1), Interaction Chemicals (Class 2), or Incidental Chemicals (Class 3). The definition of each classification is provided below.~~

~~The definition of each classification is provided below.~~

~~Tables 6.1-2 through 6.1-6 are the basic chemical inventories for the facility. Each of these tables lists a major facility structure, area, and/or system and an associated inventory of significant chemicals/chemical usage for each area. These tables do not include the listing of all incidental sludges, wastes, and waste streams which are presented in Table 6.1-1 and do not include those chemicals that have been characterized as Class 3 materials and that are not a stored "chemical". As such, those chemicals not included are not a process safety concern. Complete inventories of chemicals and chemical wastes (including incidental sludges, wastes, and waste streams) by area are provided in Chapter 2 of the Environmental Report.~~

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6.1.1.1 Chemicals of Concern (Class 1)

Chemicals of Concern (NEF 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 (CFR, 2003a) as listed below.

High Risk Chemicals of Concern

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

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

Intermediate Risk Chemicals of Concern

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 5000 times the values in Table 2 of Appendix B to 10 CFR 20 (CFR, 2003e).
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.

Non-Licensed Chemicals of Concern

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 (CFR, 2003f) – OSHA Process Safety Management
2. 40 CFR, 68 (CFR, 2003g) – EPA Risk Management Program.

These chemicals represent, based on their inherent toxic, reactive, or flammable properties, a potential for severe 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 (UF₆) is the only licensed material-related chemical of concern (NEF Class 1) that will be used at the facility. There are no non-licensed chemicals of concern at the facility. Table 6.1-1 identifies the hazards associated with UF₆, UO₂F₂, and HF; only UF₆ is considered to be process chemical. Tables 6.1-2 – 6.1-4 identify the locations and amounts of UF₆, UO₂F₂, and HF that will be present at the site.

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6.1.1.2 Interaction Chemicals (Class 2)

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

6.1.1.3 Incidental Chemicals (Class 3)

The facility will use other chemicals that are neither chemicals of concern nor interaction chemicals. Some of these incidental chemicals (NEF Class 3) include those that have the

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and Liquid UF₆. This figure shows that UF₆ expands roughly 53% going from a solid at 21°C (70°F) to a liquid at 113°C (235°F). Department of Transportation cylinder fill limits are based on UF₆ density at 121°C (250°F) and provide five percent ullage or free volume as a safety factor to prevent hydraulic rupture due to heating.

Other physical properties of UF₆ are presented in Table 6.1-65, Physical Properties of UF₆.

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6.1.2.1.2 Reactivity

UF₆ does not react with oxygen, nitrogen, carbon dioxide, or dry air, but it does react with water. For this reason, UF₆ is handled in leak tight containers and processing equipment. When UF₆ comes into contact with water, such as the water vapor in the air, the UF₆ and water react, forming HF gas and a solid uranium-oxyfluoride compound (UO₂F₂) which is commonly referred to as uranyl fluoride. Additional information on UF₆ reactions with water is provided in Section 6.2.1, Chemistry and Chemical Reactions.

UF₆ is also incompatible with a number of other chemicals including hydrocarbons and aromatics but none of these chemicals are used in or within proximity of UF₆ process systems.

6.1.2.1.3 Toxicological

If UF₆ is released to the atmosphere, the uranium compounds and HF that are formed by reaction with moisture in the air are chemically toxic. Uranium is a heavy metal that, in addition to being radioactive, can have toxic chemical effects (primarily on the kidneys) if it enters the bloodstream by means of ingestion or inhalation. HF is an extremely corrosive gas that can damage the lungs and cause death if inhaled at high enough concentrations. Additional information on the toxicological parameters used for evaluating exposure is provided in Section 6.3, Chemical Hazards Analysis.

6.1.2.1.4 Flammability

UF₆ is not flammable and does not disassociate to flammable constituents under conditions at which it will be handled at the facility.

6.1.2.2 Hydrogen Fluoride (HF) - Chemical Properties

HF is not a direct chemical of concern (NEF Class 1), however, it is one of two byproducts of concern that would be developed in the event of most accident scenarios at the facility. Understanding its properties therefore is important in evaluating chemical process conditions.

6.1.2.2.1 Physical

HF can exist as a gas or as a liquid under pressure (anhydrous HF) or as an aqueous solution of varying strengths (aqueous hydrofluoric acid). HF vapors are colorless with a pungent odor which is detectable at concentrations above 1 ppm. It is soluble in water with a release of heat.

Releases of anhydrous HF would typically fume (due to the reaction with water vapor) so that any significant release would be visible at the point of release and in the immediate vicinity.

6.2 Chemical Process Information

6.2.1.2.3 Decontamination – Citric Acid

Contaminated components (e.g., pumps, valves, piping), once they are removed from the process areas, undergo decontamination. Oily parts are washed in a hot water wash that will remove the bulk of oil including residual uranic compounds. Once the hot water wash is complete, citric acid is used to remove residual uranic fluoride compound layers that are present on the component surfaces. The reaction of the uranium compounds with the citric acid solution produces various uranyl citrate complexes. After citric acid cleansing, the decontaminated component is subject to two additional water wash/rinse cycles. The entire decontamination operation is conducted in small batches on individual components.

Decontamination of sample bottles and valves is also accomplished using citric acid.

Decontamination was evaluated in the Integrated Safety Analysis. Adequate personnel protective features are in place for safely handling decontamination chemicals and byproducts.

6.2.1.2.4 Nitrogen

~~(See 12.3 K)~~ Gaseous nitrogen is used in the UF₆ systems for purging and filling lines that have been exposed to atmosphere for any of several reasons including: connection and disconnection of cylinders, preparing lines/components for maintenance, providing an air-excluding gaseous inventory for system vacuum pumps, and filling the interstitial space of the liquid sampling autoclave (secondary containment) prior to cylinder liquefaction.

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The nitrogen system consists of a liquid nitrogen bulk storage vessel, vaporizer, liquid and gaseous nitrogen distribution lines and instrumentation. Liquid nitrogen is delivered by tanker and stored in the storage vessel.

Nitrogen is not reactive with UF₆ in any plant operational condition. Failures of the nitrogen system were evaluated in the Integrated Safety Analysis.

6.2.1.2.5 Silicone Oil

Silicone oil is used as a heat exchange medium for the heating/chilling of various cold traps and for the CTF Huber heating units. This oil is external to the UF₆ process stream in all cases and is not expected to interact with UF₆. Failures in the heating/chilling systems were evaluated in the Integrated Safety Analysis.

6.2.1.2.6 Halocarbon Refrigerants

Halocarbon refrigerants (including R23 trifluoromethane, R404A fluoromethane blend, and R507 penta/trifluoromethane) are used in individual package chillers that will provide cooling of UF₆ cylinders and/or silicon oil heat exchange media for take-off stations, CTF take-off vessel, CTF centrifuge enclosure, and cold traps. These halocarbons were selected due to good heat transfer properties, because they satisfy environmental restrictions regarding ozone depletion, and are non-flammable. All halocarbon refrigerants are external to the UF₆ process stream in all cases and are not expected to interact with UF₆. Failures in the heating/chilling systems were evaluated in the Integrated Safety Analysis.

6.2 Chemical Process Information

6.2.1.2.7 Deleted

6.2.1.2.8 Centrifuge Cooling Water

(See ~~12.3 E~~12.1.1.5.1 and 12.1.3.1) Centrifuge cooling water is provided from the Centrifuge Cooling Water Distribution System. The function of this system is to provide a supply of deionized cooling water to the cooling coils of the centrifuges. This system provides stringent control over the operating temperature of the centrifuges to enable their efficient operation. A supplemental cooling supply (plate and frame heat exchanger located in the CUB) is provided to augment the normal cooling water from the towers during extreme hot weather conditions. Additionally, since the plant will be brought online incrementally the cooling towers may not be utilized for First Cascade Online. A bypass line has been installed to isolate the cooling towers at this point and allowing the chiller units associated with the Centrifuge Cooling Water System to provide the initial cooling. When the cooling towers become available or the heat load of the enrichment plant is high enough so that the cooling towers will be necessary the Centrifuge Cooling Water System will be lined up to direct flow through the cooling towers.

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CCWS initial fill may be accomplished by using an outside source via, tanker truck rather than DI system. Hose connection with 6" isolation valve is provided for this purpose. Centrifuge cooling water is external to the UF₆ process stream in all cases and is not expected to interact with UF₆. Failures in the centrifuge cooling water distribution system were evaluated in the Integrated Safety Analysis.

6.2.1.3 UF₆ and Construction Materials

The corrosion of metallic plant components and the deterioration of non-metallic sealing materials is avoided by specifying resistant materials of construction and by controlling process fluid purity.

Direct chemical attack by the process fluid on metallic components is the result of chemical reactions. In many cases, the affinity of the process fluid for the metal produces metallic compounds, suggesting that rapid destruction of the metal would take place. This is usually prevented by the formation of a protective layer on the surface of the metal.

Deterioration of non-metallic materials is caused by exposure to process fluids and conditions. Materials used in gaskets, valves, flexible hoses, and other sealants must be sufficiently inert to have a useful service life.

UF₆ and some of its reaction products are potentially corrosive substances; particularly HF. UF₆ is a fluorinating agent that reacts with most metals. The reaction between UF₆ and metals such as nickel, copper, and aluminum produces a protective fluoride film over the metal that inhibits further reaction. These materials are therefore relatively inert to UF₆ corrosion after passivation and are suitable for UF₆ service. Aluminum is used as piping material for UF₆ systems because it is especially resistant to corrosion in the presence of UF₆. Carbon steels and stainless steels can be attacked by UF₆ at elevated temperatures but are not significantly affected by the presence of UF₆ at the operating temperatures for the facility.

Light gas impurities such as HF and air are removed from UF₆ during the purification process. Although HF is a highly corrosive substance when in solution with water as aqueous hydrofluoric acid, it contributes very little to metal corrosion when in the presence of UF₆. This is

6.3 Chemical Hazards Analysis

Levels (AEGL) 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 6.3-1, ERPG and AEGL Level Definitions.

The consequence severity limits of 10 CFR 70.61 (CFR, 2003b) has been summarized and presented in Table 6.3-2, Licensed Material Chemical Consequence Categories. The severity limits defined in this table are developed against set criteria.

The toxicity of UF_6 is due to its two hydrolysis products, HF and UO_2F_2 . The toxicological effects of UF_6 as well as these byproducts were previously described in Section 6.1.2. AEGL and NUREG-1391 values for HF and UF_6 were utilized for evaluation of chemotoxic exposure. Additionally, since the byproduct uranyl fluoride is a soluble uranium compound, the AEGL values were derived 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 ERPG and AEGL values for HF are presented in Table 6.3-3, ERPG and AEGL values for HF. The ERPG and AEGL values for UF_6 (as soluble U) are presented in Table 6.3-4, ERPG and AEGL values for Uranium Hexafluoride (as soluble U). The values from NUREG-1391 for soluble uranium are presented in Table 6.3-6, Health Effects from Intake of Soluble Uranium.

Table 6.3-5, Definition of Consequence Severity Categories, presents values for HF and UF_6 (as soluble U) from the AEGL and NUREG-1391.

6.3.2.1.1 Worker Exposure Assumptions

Any release from UF_6 systems/cylinders at the facility would predominantly consist of HF with some potential entrainment of uranic particulate. An HF release would cause a visible cloud and a pungent 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.

Two worker exposure scenarios are evaluated: (1) "consequences to the local worker" (local worker) is specific to an operator working on or operating a piece of plant equipment and unexpectedly causes a release near their vicinity (1.5 m radius), and (2) "consequences to the worker elsewhere in the room" (area worker) includes any other personnel that may be present in the room (or inadvertently enters the room) where an unanticipated release has occurred. For the purposes of evaluating worker exposure in cases where a local worker would be expected to be in the immediate proximity of a release (e.g., connect/disconnect, maintenance, etc.), the 10-minute AEGL values have been used for HF and NUREG-1391 values have been used for U. In these cases, it has been presumed that the operator will fail to recognize the in-rush of air into the vacuum system and will not begin to back away from the source of the leak until HF is present. Sufficient time is available for the worker to reliably detect and evacuate the area of concern.

The local worker is a very conservative receptor for calculating consequences. The local worker is assumed to remain in the immediate vicinity for 10 seconds following the release. Local worker exposures are evaluated using the values listed in Table 6.3-5, Definition of Consequence Severity Categories, which are 10-minute AEGL values for HF and for U. In this instance, it is conservatively presumed that the operator will not recognize the in-rush of air into

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6.3 Chemical Hazards Analysis

the vacuum system and will not begin to back away from the source of the leak until HF is present. The 10-second assumed stay time is sufficient for the local worker to detect and evacuate the area of concern.

The area worker, or "worker elsewhere in the room", are all other personnel present in the room (or inadvertently enter the room) that are not working on or operating the equipment that unexpectedly causes the release. The use of a longer (2.5 minute) exposure criteria is appropriate. These individuals may be further away from the release point and may take longer to recognize that a release has occurred and evacuate the area. Two and a half minutes is sufficient time for the worker to reliably detect and evacuate the area of concern. Area worker exposures are evaluated using the values in Table 6.3-5, which are the 10-minute AEGL values for HF and for U. This is also a conservative application of the 10-minute AEGL limit as applied to the 2.5-minute area worker stay time. For the purposes of evaluating worker exposures for workers who may be present elsewhere in the room of release, the values in Table 6.3-5, Definition of Consequence Severity Categories, which are the 10-minute AEGL values, have been used. Once a release is detected the worker is assumed to evacuate the area of concern. Sufficient time is available for the worker to reliably detect and evacuate the area of concern.

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Another assumption made in conducting consequence severity analysis is that for releases precipitated by a fire event, only public exposure was 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 generate chemical/radiological release must 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 any 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.

6.3.2.1.2 Public Exposure Assumptions

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

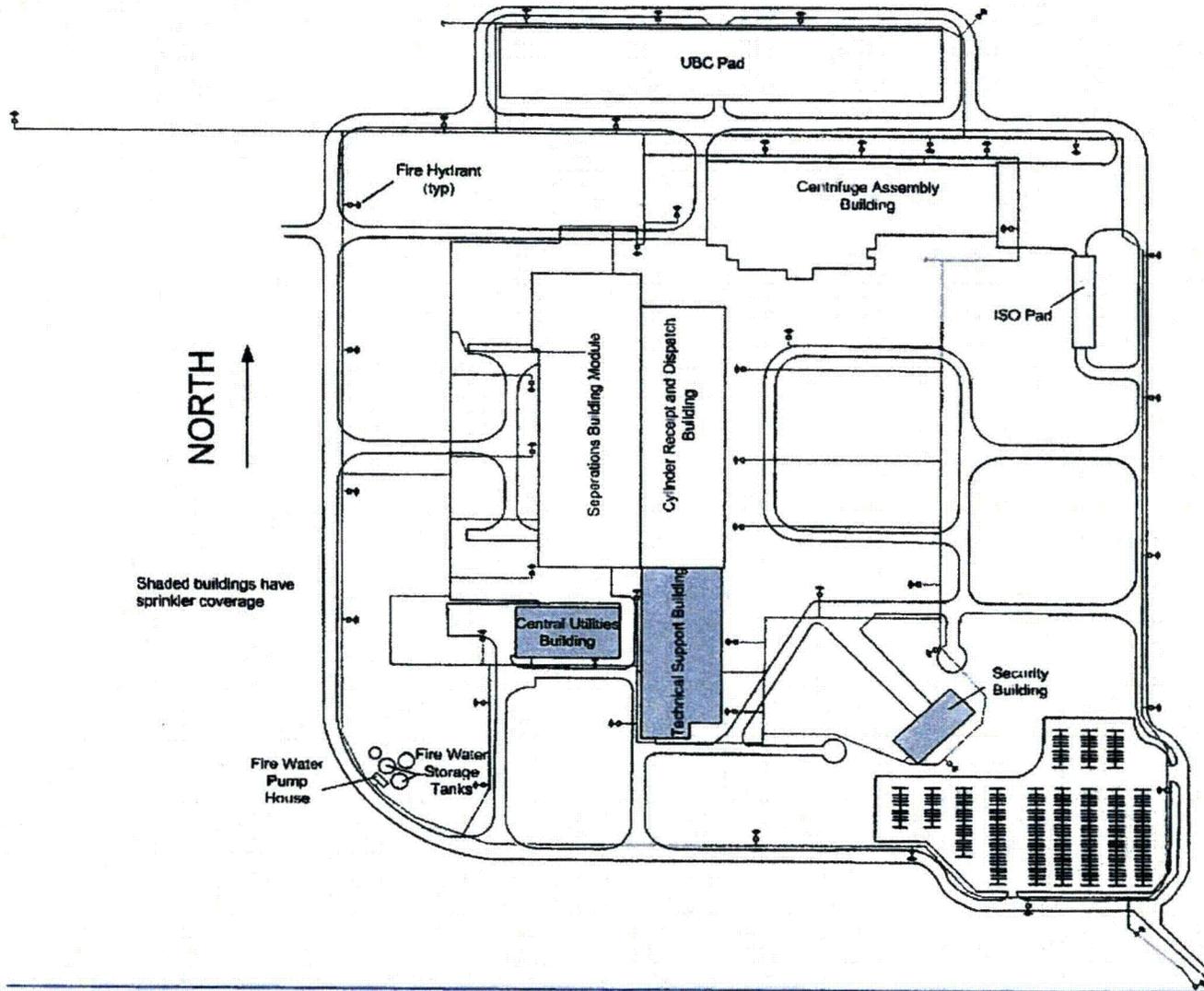
Public exposures were estimated to last for a duration of 30 minutes. This is consistent with self-protective criteria for UF_6 /HF plumes listed in NUREG-1140.

6.3.2.2 Chemical Release Scenarios

The evaluation level chemical release scenarios based on the criteria applied in the Integrated Safety Analysis are presented in the NEF Integrated Safety Analysis Summary. Information on the criteria for the development of these scenarios is also provided in the NEF Integrated Safety Analysis Summary.

6.3.2.3 Source Term

The methodologies used to determine source term are those prescribed in NUREG/CR-6410 and supporting documents. The following methodologies are approved by the U.S. Nuclear Regulatory Commission:



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Figure 7.5-1 Exterior Fire Protection System Overall Site Plan Sheet 1 of 2

10.1 Site-Specific Cost Estimate

- All areas of the plant are sectioned off into ~~Unrestricted and Restricted Areas~~ the Restricted Area and Radiologically Controlled Area (RCA). ~~Restricted Areas~~ RCAs 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. LBDCR-10-0042
- Non-radioactive process equipment and systems are minimized in locations subject to potential contamination. This limits the size of ~~the Restricted Areas~~ RCAs and limits the activities occurring inside these areas. LBDCR-10-0042
- Local air filtration is provided for areas with potential airborne contamination to preclude its spread. Fume hoods filter contaminated air in these areas.
- Curbing, pits, or other barriers are provided around tanks and components that contain liquid radioactive wastes. These serve to control the spread of contamination in case of a spill.

10.1.5.3 Worker Exposure and Waste Volume Control

The following features primarily serve to minimize worker exposure to radiation and minimize radioactive waste volumes during decontamination activities. As a result, the spread of contamination is minimized as well.

- During construction, a washable epoxy coating is applied to floors and paint is applied to walls that might be radioactively contaminated during operation. The coating will serve to lower waste volumes during decontamination and simplify the decontamination process. The coating is applied to floors and walls that might be radioactively contaminated during operation that are located in ~~the Restricted Areas~~ an RCA. LBDCR-10-0042
- Sealed, nonporous pipe insulation is used in areas likely to be contaminated. This will reduce waste volume during decommissioning.
- Ample access is provided for efficient equipment dismantling and removal of equipment that may be contaminated. This minimizes the time of worker exposure.
- Tanks are provided with accesses for entry and decontamination. Design provisions are also made to allow complete draining of the wastes contained in the tanks.
- Connections in the process systems provided for required operation and maintenance allow for thorough purging at plant shutdown. This will remove a significant portion of radioactive contamination prior to disassembly.
- Design drawings, produced for all areas of the plant, will 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 assure that workers wear proper protective equipment and limit their time in the areas.

10.1 Site-Specific Cost Estimate

Decontamination of plant components and structures will require installation of two new facilities dedicated for that purpose. Existing plant buildings, such as the Centrifuge Assembly Building, are assumed to house the facilities. These facilities will be specially designed to accommodate repetitive cleaning of thousands of centrifuges, and to serve as a general-purpose facility used primarily for cleaning larger components. The two new facilities will be the primary location for decontamination activities during the decommissioning process. The small decontamination area in the Cylinder Receipt and Dispatch Building, used during normal operation, may also handle small items at decommissioning.

Decontaminated components may be reused or sold as scrap. All equipment that is to be reused or sold as scrap will be decontaminated to a level at which further use is unrestricted. Materials that cannot be decontaminated will be disposed of in a licensed radioactive waste disposal facility. As noted earlier, 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.

Any UF_6 tails remaining on site will be removed during decommissioning. Depending on technological developments occurring prior to plant shutdown, the tails may have become marketable for further enrichment or other processes. The disposition of UF_6 tails and relevant funding provisions are discussed in Section 10.3, Tails 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.

Contaminated portions of the buildings will be decontaminated as required. Structural contamination should be limited to structures in the ~~Restricted Areas~~ RCAs. The liners and earthen covers on the facility evaporative basins are assumed to be mildly contaminated and provisions are made for appropriate disposal of these materials in the decommissioning cost estimate. Good housekeeping practices during normal operation will maintain the other areas of the site clean.

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When decontamination is complete, all areas and facilities on the site will be surveyed to verify that further decontamination is not required. Decontamination activities will continue until the entire site is demonstrated to be suitable for unrestricted use.

10.1.6.2 Decontamination Facility Construction

New facilities for decontamination can be installed in existing plant buildings to avoid unnecessary expense. Estimated time for equipment installation is approximately one year. These new facilities will be completed in time to support the dismantling and decontamination of Separations Building Module 1. These facilities are described in Section 10.1.7, Decontamination Facilities.

10.1.6.3 System Cleaning

At the end of the useful life of each Separations 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. This shutdown and purging portion of the decommissioning process is estimated to take approximately three months.

11.1 Configuration Management (CM)

11.1.2 Configuration Management Scope

Configuration Management is a cross disciplinary activity impact all elements of the QA Program include:

- Design Control
- Procurement Document Control
- Instructions, Procedures, and Drawings
- Document Control
- Control of Purchased Material, Equipment and Services
- Identification and Control Materials, Part and Components
- Control of Special Processes
- Inspection
- Test Control
- Control of Measuring and Test Equipment
- Handling, Storage, and Shipping
- Inspection, Test, and Operating Status
- Nonconforming Items
- Correction Action
- Quality Assurance Records
- Audits
- Provisions for Change

These QA elements maintain configuration management by approved processes and procedures.

11.1.3 Scope of Structures, Systems, and Components

The scope of SCCs under CM includes all IROFS identified by the integrated safety analysis of the design bases and any items which are essential to the function of the IROFS. Provisions are provided within the QAPD to control design related activities. Design documents subject to configuration management include calculations, safety analyses, design criteria, engineering drawings, system descriptions, technical documents, and specifications that establish design requirements for IROFS and items essential to the function of IROFS. Design documents are maintained under configuration management commencing with initial approval.

Drawings and specifications related to IROFS or items essential to the functions of IROFS are prepared and issued for procurement, fabrication, or construction and are placed under configuration management.

~~As the plant transitions from construction to operations~~ During construction, initial start-up, and operations, the scope of documents under configuration management broadens to include, as

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11.1 Configuration Management (CM)

- System modification documents
- Engineering documents including analyses, specifications, technical reports, and drawings.

These items are documented in approved procedures.

11.1.8 Change Control

Change control for the NEF is provided throughout the design, construction and operation phases. Change control is directed by procedures and includes an appropriate level of technical, management, and safety reviews commensurate with the risk associated with the function or operation of SSCs. Maintenance of change control during these phases is summarized below. Detail change control requirements associated with quality levels are established in the QAPD.

11.1.8.1 Design Phase

Changes to the design definition are included in the change control systematic review process. Changes to the design are reviewed for 10 CFR 70.72 impacts through an Integrated Safety Analysis process. This process includes a systematic review of the design bases for consistency with LBDs. Changes that affect design or operation of IROFS are reviewed, and approved prior to implementation.

The configuration management process includes interdisciplinary reviews which ensure design changes either (1) do not impact the ISA, (2) are accounted for in subsequent changes to the ISA, or (3) are not approved or implemented.

11.1.8.2 Construction Phase

During the construction phase, changes to documents issued for construction, fabrication, and procurement will be documented, reviewed, approved, and posted in conjunction with design documents. Vendor drawings and data undergo an interdisciplinary review to ensure compliance with procurement specifications and drawings, and to incorporate interface requirements into facility documents.

During construction, design changes will be evaluated against the approved design bases. A systematic process will be used to evaluate changes in the design against the design bases of IROFS and the ISA. The configuration change process will implement the provisions of 10 CFR 70.72 (CFR, 2003e), including reporting of changes made without prior NRC approval as required by 10 CFR 70.72(d)(2) and (3). Any change that requires Commission approval, will be submitted as a license amendment request as required by 10 CFR 70.72(d)(1) and the change will not be implemented without prior NRC approval.

11.1.8.3 Operations Phase

During the operations phase and while transitioning between construction and operation, changes to design will be documented, reviewed, and approved prior to implementation. These processes implements the provisions of 10 CFR 70.72 (CFR, 2003e). Measures are provided to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties.

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12.0 PHASED OPERATION

12.0 PHASED OPERATION

~~Note: Section 12.0 is a proposed plan for proceeding with Phased Operation. Some of the information delineated below may not yet be approved for implementation. However, as specific design details becomes available and prior to operation, it they will be evaluated and approved in accordance with the Configuration Management process. Approval documentation will be clearly noted (e.g., "Approved per CG EG 2009 9999").~~

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The initial startup of the National Enrichment Facility does not include all facilities, systems, processes, and IROFS described in ISA Summary § 3.3 through § 3.8. The startup of the facility is performed in a phased approach to begin operation as soon as the required facilities, systems, processes, and IROFS are operational to support Initial Plant Operation (IPO). As delineated in SAR § 2.1.4, Transition from Design and Construction to Operations, LES is responsible for the design, quality assurance, construction, testing, initial startup, and operation of the facility. As the construction of systems is completed, the systems will undergo acceptance testing as required by procedure, followed by turnover from the construction organization to the operations organization by means of a Commissioning Acceptance Plan.

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The facility will operate in a series of phases determined by operational requirements. Initial Plant Operation (IPO) phase will include all safety systems necessary to safely conduct enrichment operations. The following phases (Production Phase 1 and Production Phase 2) will add support systems as necessary as the production capacity expands. These phases are described as follows:

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- ~~1. Initial Plant Operation During the initial phase of operation at the NEF, all Structures, Systems, and Components (SSCs) that are required to support the start up and early operation of the enrichment facility will be are completed and brought online as necessary to support that function. UF₆ operations will be are conducted in SBM 1001. This building will contain all the required SSCs to support Initial Plant Operations. Additional support functions will be are brought into operation in the following phases. In addition to the permanent plant equipment, some temporary systems will be are installed in the UF₆ Handling Area of SBM 1001 to support operations in place of systems that will not be are not completed. These systems include a storage area for a small amount of of radioactive material or solid and liquid waste, and any contaminated equipment that requires storage in preparation for decontamination and repairs, waste treatment, or disposal. In addition, the Local Extract GEVS function will be is combined with the Pumped Extract GEVS to support Initial Plant Operations.~~

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The cascades in SBM 1001 will be are brought online in modules that contain all the systems that are necessary to support the function of the individual cascades. For instance, when the first cascade is started up, enough feed, product, and tails stations will be are online to support operation of that cascade. Subsequent cascade modules will be are incrementally brought online as they are needed.

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Other support systems not directly part of the UF₆ enrichment process will be are contracted. For example, Laboratory analysis of UF₆ material will be is contracted to a certified analytical laboratory, and a temporary personnel decontamination trailer will be provided on site.

- ~~2. Production Phase 1 When enough product has been is enriched and is ready to be shipped to a customer, several other support functions will be completed are available and~~

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12.0 PHASED OPERATION

ready to support plant operations. These support functions include cylinder storage and sampling:

- ~~▲ Cylinder Storage Facilities (South CRDB)~~
- ~~▲ GEVS systems (Local Extract GEVS, Fume Hood GEVS)~~
- ~~▲ Liquid Sampling Autoclave~~
- ~~▲ Ventilated Room~~
- ~~▲ Sub-Sampling System~~
- ~~▲ UBC Basin~~
- ~~▲ UBC Storage Pad~~
- ~~▲ Product Donor and Receiver Blending Stations~~
- ~~▲ Personnel Decontamination Shower~~
- ~~▲ Chemistry Lab~~
- ~~▲ Mass Spectrometry Lab~~
- ~~▲ Centrifuge Cooling Water Towers~~

Addition of these SSCs will provide several additional functions that will support commercial production and shipment to customers. They include This provides additional cylinder storage, the ability to sample product prior to shipment, and other chemistry activities.

3. ~~Production Phase 2~~ At the completion of this phase, functions supporting sample analysis, wet and dry waste collection and treatment, and radioactive decontamination and maintenance of plant equipment will be are available. SSCs include:

- ~~▲ Permanent Cylinder Receipt/Shipment~~
- ~~▲ Expanded Cylinder Storage in CRDB~~
- ~~▲ Liquid Effluent Collection and Treatment~~
- ~~▲ Solid Waste Collection~~
- ~~▲ Vacuum Pump Rebuild Workshop~~
- ~~▲ Decontamination Workshop~~

With the addition of these final SSCs, the NEF plant will be fully functional. Additional cascades and support equipment will be can be added in the future to increase production, but the plant will be is fully capable of carrying out continuous commercial production at this point.

4. ~~Production Phase 3~~ When construction activities support, Cascade modules in Cascade Hall 1002 will be are started up incrementally as needed to support continued plant expansion. This incremental start up will continues until Cascade Halls 1001 and 1002 are both fully operational.

5. ~~Production Phase 4~~ As SSCs are ready for operation, the extension of SBM 1001 will be brought online using the same modular approach used to start up all previous cascades.

~~Operate While Constructing~~

An Operate While Constructing program is necessary to implement controls for continued construction during facility operation. The Operate While Constructing program is necessary until all cascades and expansion modifications are implemented and accepted by Operations.

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12.0 PHASED OPERATION

Operate While Constructing is a process that implements controls to ensure that the Integrated Safety Analysis for the National Enrichment Facility remains valid during operations when part of the facility is still being constructed. The process of Phased Operation, placing cascades on-line and facility expansion is estimated to take several years; therefore, Operate While Constructing is an essential safety process for the operation of the National Enrichment Facility.

The following sections provide a description of the operations that differ between final operation of the facility and the interim operation for the Phased Operation approach. Applicable portions of SAR Chapter 12 are referenced by all other LBDs impacted by the Phased Operation approach.

The following general Accident Sequences and associated IROFS are applicable to all areas containing UF₆. Because the CRDB/UBC Storage Pad are not operational and contain no UF₆, these accident sequences are not applicable to any room in the CRDB or to the UBC Storage Pad:

<u>General Accident Sequences</u>	
<u>• EE-SEISMIC-WORKER EVAC</u>	<u>IROFS39a</u>
<u>• FF-WORKER EVAC</u>	<u>IROFS36a, 36d, & 36i, IROFS39b</u>
<u>• EE-CHEM RELEASE-WORKER EVAC</u>	<u>IROFS39c</u>
<u>• EE-TORNADO MISSILE-SBM-CRDB SHELL & BUNKER WORKER</u>	<u>IROFS39d</u>

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12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

~~12.112.1.1 (1.1.2) Facility Differences for Phased Initial Plant Operations (IPO)~~

~~The differences between the facility as described in §§ 1.1.2, 1.1.4, 4.0.1, and 4.0.2 and the facility at the start of Initial Plant Operations through Production Phase 2 are described below. Phased Operation does not impact Safety Analysis Report §§ 2.0, 3.0, 5.0, and 7.0 through 11.0.~~

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~~12.1.112.1.1.1 (1.1.2) Separations Building Modules~~

~~12.1.1.1.1 Cascade System~~

~~(Approved per CC-LS-2010-0001) At the beginning of Initial Plant Operations, only one cascade within SBM-1001 will be is operational. Additional Cascades will be are brought into service as they are commissioned. The Cascade System is operational as described in ISA Summary § 3.4.3 with the exception that only one cascade module within SBM1001 is operational at the beginning of IPO. A cascade module of Feed, Product, and Tails Stations, UF₆ gas transport equipment (piping, valves, centrifuges), and the Contingency Dump System piping and components for that cascade module. Additional cascade modules are brought into service as they are commissioned. NRC approval is required prior to introduction of UF₆ into each cascade module. At the end of IPO, cascade modules 1 through 6 are operable.~~

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~~Accident Sequence EE-SEISMIC-SBM and associated IROFS27e and IROFS41 are applicable (IROFS28 is not applicable).~~

~~12.1.1.1.2 Process Services Corridor (PCS)~~

~~(Approved per CC-LS-2010-0001) The Process Services Corridor (PSC) for SBM-1001 will be operational, but will lack gas transport equipment for cascades that are not on line (NaF Traps, Pump and Trap Sets, process headers, etc). This equipment is installed and operated as additional cascades are completed through Initial Plant Operations, Production Phase 1, and Production Phase 2.~~

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~~12.1.1.1.3 UF₆ Solid Feed System The UF₆ Handling Area, including the Blending and Liquid Sampling Area, will have has the following differences:~~

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~~The UF₆ Solid Feed System and Feed Purification Subsystem are operational as described in ISA Summary § 3.4.2 except a minimum of three (3) Solid Feed Stations (SFS) and one (1) Feed Purification Low Temperature Take-off Station (LTTS) are required to be operable for FCOL enrichment operations. As IPO progresses, additional stations are completed and brought online as needed to support the incremental start up on cascades. The second Feed Purification Station (if operable) and all operable SFS not in use for enrichment operations contain a full feed cylinder at the beginning of IPO.~~

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~~Accident sequences UF1-1, UF2-1, and associated IROFS4 and 5 are applicable.~~

~~1. (Approved per CC-LS-2009-0002, Rev. 1) The UF₆ Solid Feed Stations, Feed Purification Stations, Product Take-off Stations, and Tails Take-Off Stations associated with SBM-1001 will be are installed and brought online as needed to support starting up cascades in SBM-1001.~~

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12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

12.1.1.1.4 Product Low Temperature Take-off System

~~2. (Approved per CC-LS 2009-0002, Rev. 1) The Blending Receiving and Donor Stations are not available for Initial Plant Operation. Blending operations are not conducted until Production Phase 1. (Approved per CC-LS 2009-0002, Rev. 1) The Autoclaves, Product Blending Donor Stations, and Product Blending Receiver Stations will not be installed until Production Phase 1. Without these components, no product cylinders can be shipped off site. The Product Low Temperature Take-off System is operational as described in ISA Summary § 3.4.4 except a minimum of three (3) Product LLTS are required to be operable for FCOL enrichment operations. As IPO progresses, additional Product LLTS are brought online as needed to support the incremental start up of cascades. All operable Product LLTS not in use for enrichment operations contain an empty Product Cylinder at the beginning of IPO.~~

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Accident sequences PT2-1 and PT2-2 and associated IROFS1 and IROFS2 are applicable.

12.1.1.1.5 Tails Low Temperature Take-off System (LLTS)

~~(Approved per CC-LS 2009-0002, Rev. 1) The Autoclaves are installed at the time of Production Phase 1. Without these components, product cylinders can not be shipped to customers but can be shipped off site for temporary storage. The Tails LLTS is operational as described in ISA Summary § 3.4.5 except a minimum of three (3) Tails LLTS are required to be operable for FCOL enrichment operations. As IPO progresses, additional Tails LLTS are brought online as needed to support the incremental start up of cascades. All operational stations not in use for enrichment operations contain a full feed cylinder at the beginning of IPO.~~

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10-0002

Accident sequence TT2-1 and associated IROFS1 and 2 are applicable.

12.1.1.1.6 Blending System

The Blending Receiving and Donor Stations are not needed for IPO. However, the Blending Donor and Receiver Stations are operable for storage of empty or full product cylinders.

Accident sequences PB1-1 and PB2-1 and associated IROFS1,2,4, and 5 are applicable.

12.1.1.1.7 Product Liquid Sampling System

The Product Liquid Sampling System autoclaves are not available and not needed for IPO. Without these components, product cylinders can not be shipped to customers but can be shipped off site for temporary storage.

LBDCR-
10-0033

Because the autoclaves are not available, accident sequences PB4-1, PB4-2, PB4-3, PB4-4, EE-TORNADO MISSILE-SBM PUBLIC, and EE-SEISMIC-SBM and associated IROFS10, 11, 12, and 28 are not applicable. Note that the seismic events are applicable to the SBM but the autoclave contribution to the total release is not applicable.

- ~~1. The Blending and Liquid Sampling Area has been moved into SBM 1001 UF₆ Handling Area. The Blending Receiving and Donor Stations and Liquid Sampling Autoclaves will not be available for Initial Plant Operation. Blending and liquid sampling will not be conducted until Production Phase 1.~~

12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

12.1.1.1.8 Rail Transporter

~~(CC-OP-2009-0002 Pending)~~ The Rail Transporter will travel on rails embedded in the floor of the UF₆ Handling Area. These rails run the entire width of the module; east to the CRDB and to the west through doors onto a concrete pad where cylinders will be delivered during Initial Plant Operations and Production Phase 1 Operations IPO. The rail runs east to the CRDB. Upon commencement of Production Phase 2, cylinders will be delivered through the CRDB, and the west entrance of the UF₆ Handling Area will no longer be used for cylinder deliveries.

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There is no accident sequence or IROFS directly associated with the Rail Transporter.

12.1.1.1.9 Inventory Weighing

~~(Approved per CC-EG-2009-0291)~~ A weigh station will be located in the UF₆ Handling Area for Initial Plant Operations. Upon commencement of Production Phase 1 Operations, the weigh scale in the CRDB will be functional and the one in the SBM will be removed. Inventory weighing is performed for each cylinder that enters or exits the SBM during IPO using a temporary scale in the UF₆ Handling Area of SBM1001. The scale is identical to the scales described in ISA Summary § 3-4-11-1-2 C. The temporary weigh scale is capable of weighing a load of 17 MT (37,500 lb) with a tolerance of ±2.5 kg (±5.5 lb) and capable of accepting a load of up to 20 MT (44,100 lb). The scale has reader and printout facilities.

LBDCR-10-0002

There is no accident sequence or IROFS directly associated with the weigh scales.

12.1.1.1.10 Gaseous Effluent Ventilation System (GEVS)

The Gaseous Effluent Ventilation System (GEVS) is constructed as two separate systems. Pumped Extract GEVS and CRDB GEVS. Pumped Extract GEVS is permanently installed in the UF₆ Handling Area of SBM1001 and is operational for IPO. The local extract ductwork that is used in the SBM is temporarily connected to the Pumped Extract GEVS to support IPO.

All GEVS accident sequences (CL3-1, CL3-2, CL3-3, VR1-1, VR1-2, VR 2-2, and FF25-2) and associated IROFS (IROFS20, 21, 24a, 24b, and 37) are for CRDB operations and therefore not applicable to IPO.

LBDCR-10-0033

Accident sequence LOSS OF SAFE-BY-DESIGN ATTRIBUTE is applicable for the Pumped Extract GEVS.

There is no accident sequence or IROFS directly associated with the local extract function of the CRDB GEVS.

12.1.1.1.11 Radiation Monitoring Control Room

The Radiation Monitoring Control Room is not operational. Normal ingress and egress from the enrichment processing areas is through the controlled SBM entrance. A radiological control point is established within the SBM designed to be the point of demarcation between non-contaminated areas and potentially contaminated areas of the facility. Personnel contamination detection equipment is staged at the control point. There is a personnel decontamination facility containing hand washing capabilities and safety showers adjacent to the SBM.

12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

There is no accident sequence or IROFS directly associated with the Radiation Monitoring Control Room.

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12.1.212.1.1.2 (4.1.2) Technical Services Building (TSB)

~~E. The Control Room will be operational as described in ISA Summary § 3.3.1.2.2.1.1.2.~~

LBDCR-10-0002

~~F. The Training and Simulator Rooms will not be are operational as described in ISA Summary § 3.3.1.2.2.2. The PCS Training software will be temporarily installed in a classified trailer to facilitate Operator Training in preparation for Initial Plant Operations.~~

~~G. The Central Alarm Station (CAS) Area will be is operational as described in ISA Summary § 3.3.1.2.2.3.~~

12.1.1.2.1 Medical Room

~~D. (Approved per CC-LS-2009-0002, Rev. 2) The Medical Room will be is operational for general first aid cases. Injuries requiring more than general first aid will ~~be~~ are transported off site to local area medical facilities.~~

LBDCR-10-0002

There is no accident sequence or IROFS directly associated with the Medical Room.

~~E. The Emergency Operations Center Room will be is operational as described in ISA Summary § 3.3.1.2.2.5.~~

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~~F. The Technical Support Center Assembly Room will be is operational as described in ISA Summary § 3.3.1.2.2.6.~~

LBDCR-10-0033

12.1.1.2.2 Break Room

~~G. (Approved per CC-LS-2009-0002, Rev. 2) The Break Room will is not be operational.~~

LBDCR-10-0002

There is no accident sequence or IROFS directly associated with the Break Room.

12.1.1.2.3 I&C Electrical Shop

~~H. (Approved per CC-LS-2009-0002, Rev. 2) The I&C Electrical Shop Room will is not be operational. The I&C Electrical Shop serves as a work area for general electrical and I&C components and maintenance. Maintenance on non-contaminated equipment will ~~be~~ is delayed until the I&C Electrical Shop is available or is conducted in other locations on-site or off-site, as necessary, based on the equipment and maintenance required.~~

LBDCR-10-0002

There is no accident sequence or IROFS directly associated with the I&C Electrical Shop.

12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

12.1.1.2.4 Mechanical Shop

~~I. (Approved per CC-LS-2009-0002, Rev. 2) The Mechanical Shop Room will is not be operational. The Mechanical Shop serves as a work area for general mechanical maintenance and work such as painting or welding. Maintenance on non-contaminated equipment will be is delayed until the Mechanical Shop is available or conducted in other locations on-site or off-site, as necessary, based on the equipment and maintenance required.~~

LBDCR-10-0033

LBDCR-10-0002

There is no accident sequence or IROFS directly associated with the Mechanical Shop.

LBDCR-10-0002

~~J. The Chemical Storage Room will be is operational as described in ISA Summary §3.3.1.2.2.10.~~

12.1.1.2.5 Waste Processing Room

~~A. (Approved per CC-LS-2010-0001) The Waste Processing Room will be operational. The Waste Processing Room is not operational. The Waste Processing Room serves as a processing-handling area effor non-radioactive wastes. Non-radioactive wastes will are is either be stored under appropriate safety controls until processing-handling systems are available, or shipped off-site to a processing facility for treatment and/or disposal at a licensed facility.~~

LBDCR-10-0033

LBDCR-10-0002

There is no accident sequence or IROFS directly associated with the Waste Processing Room.

12.1.1.2.6 Environmental Monitoring Laboratory

~~B. (Approved per CC-LS-2009-0002, Rev. 1) The Environmental Monitoring Laboratory will is not be operational. Instead, samples will be are collected and shipped to a certified testing facility for analysis. The sample containers will are not be returned to LES, but will be are disposed of by the receiving facility.~~

LBDCR-10-0002

There is no accident sequence or IROFS directly associated with the Environmental Monitoring Laboratory

~~12.1.3~~ 12.1.1.3 (1.1.2) Cylinder Receipt and Dispatch Building (CRDB)

12.1.1.3.1 Solid Waste Collection Room

~~(1.1.4) (Approved per CC-LS-2009-0002, Rev. 1) The Solid Waste Collection Room will not be is not operational. The Solid Waste Collection Room is designed to preceespackage both wet and dry low-level radioactive solid waste.~~

LBDCR-10-0002

12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

The small quantity of solid waste that is expected to be generated during IPO at NEF will be is stored in accordance with appropriate radiological and criticality safety controls until the Solid Waste Collection Room is completed. placed in a lined 55 gal drum with <300 g U²³⁵ as determined through bookkeeping. Once the drums have been filled they are sealed with a tamper-indicating device (TID) and placed into the Material Control and Accountability (MC&A) item control program. Up to four drums are stored in the Ventilated Storage Room in the UF₆ Handling Area in SBM1001. A qualified contracted company conducts non-destructive assay (NDA) on the drums to determine the final U²³⁵ content. Once the assay is complete Radiation Protection and MC&A Departments can release the drums from the MC&A item control inventory to radioactive material storage areas (RMAs) external to the SBM. The drums will remain in storage until either further evaluation by radiation protection free releases the material or sufficient quantity is accumulated to prepare an offsite shipment.

Because the Solid Waste Collection Room is not completed, accident sequences SW1-1 and SW1-2 and associated IROFS14a and IROFS 14b are not applicable.

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Transitional accident sequences TVR1-1, TVR1-2, and TVR1-3 have been identified that require implementation of existing IROFS14a and 14b, and IROFS31a, 31b, and 31c to the Ventilated Storage Room. See ISA Summary Table 4-4, Transitional Accident Sequence and Risk Index, and 4-5, Transitional Accident Sequence Descriptions.

12.1.1.3.2 Vacuum Pump Rebuild Workshop

~~(Approved per CC-LS-2009-0002, Rev. 1)~~ The Vacuum Pump Rebuild Workshop will not be is not operational. The rebuilding of vacuum pumps is a planned evolution. In the unlikely event that a rebuild of a vacuum pump containing UF₆ is required, the pump will be is replaced with a clean vacuum pump and the contaminated pump stored in accordance with appropriate radiological controls until the Vacuum Pump Rebuild Workshop is completed.

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There is no accident sequence or IORFS directly associated with the Vacuum Pump Rebuild Workshop.

12.1.1.3.3 Decontamination Workshop

~~15(4.9.1) (Approved per CC-LS-2009-0002, Rev. 1)~~ The Decontamination Workshop will not be is not operational. The decontamination systems in this workshop are designed for radioactive decontamination of materials and equipment used in uranium hexafluoride systems, waste handling systems, and other areas of the plant. The small quantity of contaminated equipment that is expected will be is stored in accordance with appropriate chemical, radiological, and criticality safety controls until the Decontamination Workshop is completed or shipped off site to a processing facility for treatment and/or disposal at a licensed facility.

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12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

Equipment, other than pumps, requiring radioactive decontamination is placed in a lined 55 gal drum with <300 g U²³⁵ as determined through bookkeeping. Once the drums have been filled they are sealed with a tamper-indicating device (TID) and placed into the Material Control and Accountability (MC&A) item control program. Up to four drums are stored in the Ventilated Storage Room in the UF₆ Handling Area in SBM1001. A qualified contracted company conducts non-destructive assay (NDA) on the drums to determine the final U²³⁵ content. Once the assay is complete Radiation Protection and MC&A Departments can release the drums from the MC&A item control inventory to radioactive material storage area (RMAs) external to the SBM. The drums will remain in storage until either further evaluation by radiation protection free releases the material or sufficient quantity is accumulated to prepare an offsite shipment.

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Pumps requiring decontamination will be stored in place until the decontamination workshop is completed and running.

Because the Decontamination Workshop is not completed, accident sequence LOSS OF SAFE-BY-DESIGN ATTRIBUTE is not applicable.

12.1.1.3.4 Ventilated Room

~~16(Approved per CC-EG-2009-0369 Pending)~~ The Ventilated Room will not be is not operational. The main activities carried out in the Ventilated Room are servicing chemical traps by removing spent carbon, aluminum oxide, and sodium fluoride and replacing damaged and leaking valves on cylinders which contain UF₆. Servicing chemical traps is a planned evolution and will not be is not required or planned before Ventilated Room is completed. ~~A temporary room has been constructed will be is constructed in the UF₆ Handling Area in SBM 1001 for the purpose of storage of any contaminated equipment or waste generated during Initial Plant Operations. This room will be is connected to the Pumped Extract GEVS, which is also located in UF₆ Handling Area. The room will be is used for storage only; no processing of equipment or materials will be are conducted. Although a leaking valve on a cylinder containing UF₆ is not expected, if one is identified, the potential leakage will be is stopped using appropriate procedural guidance and the cylinder stored in an appropriate (feed or product) station until repairs can be conducted or the cylinder can be returned to the vendor.~~

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Because the Ventilated Room is not available, accident sequences VR1-1, VR1-2, VR1-3, VR1-5, VR2-1, VR2-2, VR2-7, FF24-1, FF25-1, and FF25-2 and associated IROFS3,21,23a, 23b, 24a, 35, 36d, 37, 47b and accident sequence LOSS OF SAFE-BY-DESIGN ATTRIBUTE are no Applicable for the Ventilated Room Transitional accident sequences associated with this room have been identified that require implementation of existing IROFS. See ISA Summary Tables 4-4 and 4-5.

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~~12.1.1.3.5 Reserved (CC-EG-2010-0005 Pending)(Approved per CC-EG-2010-0005 Rev. 1)~~ The Liquid Effluent Collection and Treatment Room will not be is not operational. Instead, the various types of aqueous and non-aqueous liquid wastes generated by plant operations and processes in the facility will be are collected and either shipped off site to an appropriate treatment and disposal facility or stored on site in accordance with appropriate radiological and criticality safety controls until the Liquid Effluent Collection and Treatment Facility is completed.

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(Approved by CC-EG-2008-0519) The LECTS Room will also be used for trap filling. Until the LECTS Room is available, clean (non-UF₆ contaminated) trap fill operations (carbon, aluminum oxide, and NaF) will be are conducted in the Trap Filling and Vacuum Pump Building (2300). Building 2300 will also be used for chemical trap

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12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

~~drying; vacuum pump receipt inspections; PFPE oil sampling; PFPE oil analysis; and helium leak testing. No licensed materials will be are contained in this building. Contaminated traps will not be reused.~~

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12.1.1.3.6 Contaminated Material Handling Room

~~17(4.9.2) (Approved by CC-RW-2009-0001) The Contaminated Material Handling Room will not be is not operational. Instead, contaminated disposable protective clothing will be is collected, monitored and either shipped off site to a licensed disposal facility or stored on site in accordance with appropriate controls until the Contaminated Material Handling Room and Solid Waste Collection Room are completed and implemented.~~

Radioactive waste is placed in a lined 55 gal drum with < 300 g U²³⁵ as determined through bookkeeping. Once the drums have been filled they are sealed with a tamper-indicating device (TID) and placed into the Material Control and Accountability (MC&A) item control program. Up to four drums are stored in the Ventilated Storage Room in the UF₆ Handling Area in SBM1001. A qualified contracted company conducts non-destructive assay (NDA) on the drums to determine the final U²³⁵ content. Once the assay is complete Radiation Protection and MC&A Departments can release the drums from the MC&A item control inventory to radioactive material storage areas (RMAs) external to the SBM. The drums will remain in storage until either further evaluation by radiation protection free releases the material or sufficient quantity is accumulated to prepare an offsite shipment.

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Equipment, other than pumps, requiring radioactive decontamination is treated the same as radioactive waste (described above) except that it is stored until decontamination facilities are available on site. Pumps requiring decontamination will be stored in place until the decontamination workshop is completed and running.

There is no accident sequence or IROFS directly associated with the Contaminated Material Handling Room.

12.1.1.3.7 Gaseous Effluent Ventilation System (GEVS)

~~Approved by CC-EG-2009-0293) The Gaseous Effluent Ventilation System (GEVS) Room is not operational. The GEVS System is constructed as three separate systems, Pumped Extract GEVS, Local Extract GEVS, and Fume Hood GEVS (Fume Hood GEVS is not required for Initial Plant Operations). Pumped Extract GEVS is permanently installed in the UF₆ Handling Area of SBM 1001 and is operational for Initial Plant Operations. The Pumped Extract GEVS is temporarily connected to Local Extract ductwork in the SBM to support Initial Plant Operations.~~

(CC-EG-2009-0101 Pending) When the GEVS Room is complete, the permanent Local Extract GEVS will be installed along with the Fume Hood GEVS in that room. Once these GEVS systems are operational, the Pumped Extract GEVS' temporary connection to the Local Extract ductwork will be isolated. The Gaseous Effluent Ventilation System (GEVS) is constructed as two separate systems, Pumped Extract GEVS and CRDB GEVS. Pumped Extract GEVS is permanently installed in the UF₆ Handling Area of SBM1001 and is operational for IPO. The local extract ductwork that is used in the SBM is temporarily connected to the Pumped Extract GEVS to support IPO.

12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

Because of this temporary cross-connection, there are limitations to the local extract capability. The following measures are in place to ensure adequate flow is provided at each local extract station:

- Only two local extract flexible hose stations are allowed to be open at any one time (IF the Ventilated Storage Room is online, THEN only one flexible hose station is allowed to be in use).
- Configuration control is maintained by the Shift Manager and caution tags on the local extract flexible hose station isolation valves.

All GEVS accident sequences (CL3-1, CL3-2, CL3-3, VR1-1, VR1-2, VR2-2, and FF25-2) and associated IROFS (IROFS20, 21, 24a, 24b, and 37) are for CRDB operations and therefore not applicable to IPO.

Accident sequence LOSS OF SAFE-BY-DESIGN ATTRIBUTE is applicable for the Pumped Extract GEVS.

There is no accident sequence or IROFS directly associated with the local extract function of the CRDB GEVS.

12.1.1.3.8 Mass Spectrometry Laboratory

~~G. The Gaseous Effluent Ventilation System (GEVS) Room will not be operational. The GEVS System will be constructed as three separate systems, Pumped Extract GEVS, Local Extract GEVS, and Fume Hood GEVS (Fume Hood GEVS is not required for Initial Plant Operations).~~

- ~~a. Pumped Extract GEVS will be permanently installed in the UF₆ Handling Area of SBM 1001 and will be operational for Initial Plant Operations. The Pumped Extract GEVS will be temporarily connected to Local Extract ductwork in the SBM to support Initial Plant Operations.~~
- ~~b. When the GEVS Room is complete, the permanent Local Extract GEVS will be installed along with the Fume Hood GEVS in that room. Once these GEVS systems are operational, the Pumped Extract GEVS' temporary connection to the Local Extract ductwork will be isolated.~~

~~19 (Approved per CC-LS-2009-0002, Rev. 1) The Mass Spectrometry Laboratory will not be operational. Instead, samples will be collected and shipped to a certified testing facility for analysis. Contaminated sample containers will not be returned to LES, but will be disposed of by the receiving facility.~~

Because the Mass Spectrometry Laboratory is not completed, accident sequence LOSS OF SAFE-BY-DESIGN ATTRIBUTE is not applicable.

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12.1.1.3.9 Chemical Laboratory

~~20(Approved per CC-LS-2009-0002, Rev. 1)~~ The Chemical Laboratory ~~will not be~~ is not operational. Instead, samples ~~will be~~ are collected and shipped to a certified testing facility for analysis. Contaminated sample containers ~~will not be~~ are not returned to LES, but ~~will be~~ are disposed of by the facility.

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Because the Chemical Laboratory is not completed, accident sequences CL3-1, CL3-2, and CL3-3 and associated IROFS24b, 43, and 46 and LOSS OF SAFE-BY-DESIGN ATTRIBUTE are not applicable.

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~~J.~~12.1.1.3.10 Radiation Monitoring Control Room

The Radiation Monitoring Laboratory is not operation. Instead, samples are collected and shipped to a certified testing facility for analysis.

There is no accident sequence or IROFS directly associated with the Radiation Monitoring Laboratory~~(Approved per CC-LS-2010-0001)~~ The Radiation Monitoring Control Room will not be ~~is not operational. Normal ingress and egress from the enrichment processing areas will be~~ is through the ~~controlled SBM west entrance. A radiological control point is established within the SBM designed to be the point of demarcation between non-contaminated areas and potentially contaminated areas of the facility. Personnel contamination detection equipment is staged at the control point. There is a personnel decontamination facility containing hand washing capabilities and safety showers adjacent to the SBM~~ The required radiological equipment will be available.

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~~K.~~12.1.1.3.11 The Truck Bay/Shipping and Receiving Area will not be is not operational.

~~(CC-OP-2009-0002 Pending)~~ Commercial transport tractors are disconnected from the trailers carrying containers and connected to LES yard tractors which comply with IROFS36c (i.e., diesel fuel capacity less than 280 L (74 gal)). The yard tractor Transport trucks will deliver UF₆ cylinders (i.e., full 48Y feed cylinders, new or cleaned 30B product cylinders, and empty 48Y tails cylinders) to the Vehicle Loading and Unloading Area ~~a cement pad on the west side of SBM-1001 in the southwest corner.~~

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Cylinders are unloaded with a gantry crane at the Vehicle Loading and Unloading Area of SBM1001 sufficient capacity to perform all required cylinder movements. The gantry crane will lifts and transfers the cylinder to the rail transporter that sits on rails extended outside the SBM into the Vehicle Loading and Unloading Area. On completion of receipt inspection, the rail transporter will move the cylinder inside the UF₆ Handling Area. ~~The unloading of cylinders will be performed with a mobile crane of sufficient capacity to unload cylinders from the delivery vehicles directly onto the rail transporter. Cylinders are removed from the facility in the same fashion.~~

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There is no accident sequence or IROFS directly associated with the Gantry Crane at the SBM Vehicle Loading and Unloading Area.

12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

12.1.1.3.12 ~~The Cylinder Storage Areas in the CRDB will not be is not operational.~~

- ~~(Approved per CC-LS-2009-0002, Rev. 1) The buffer storage of Full feed cylinders will be is are stored in the UF₆ Handling Area in available Solid Feed, Feed Purification, and Tails, and Feed Purification Stations until the UBC Storage Pad or the South end of the CRDB are is ready to accept cylinders for storage.~~

Accident sequences UF1-1, UF2-1, and TT2-1, and associated IROFS1,2,4, and 5 are applicable.

- ~~(Approved per CC-LS-2009-0002, Rev. 2) Full product cylinders are stored in available storage will be is accomplished in the UF₆ Handling Area in Product Take-off Stations and Blending Donor and Take-Off Stations until the CRDB is ready to accept cylinders for storage for approximately 3 months after initial plant operations commence. When the autoclaves are operational, the product cylinders are liquid sampled and shipped to clients until the UBC Storage Pad or the South end of the CRDB is ready to accept cylinders for storage.~~

Accident sequences PT2-1, PT2-2, PB1-1, PB2-1, PB2-2, and CP1-2 and associated IROFS1,2,4,5, and 16a are applicable.

- ~~(Approved per CC-LS-2009-0002, Rev. 1) Full tails cylinders will be are stored in the available Tails Take-off Stations until the UBC Storage Pad or the South end of the CRDB is ready to accept cylinders for storage.~~

Accident sequence TT2-1, and associated IROFS1 and 2 are applicable.

23.0.1 ~~Centrifuge Assembly Building (CAB)~~

~~The CAB will be is operational as described in ISA Summary § 3.3.1.4.~~

23.0.2 ~~Not Used~~

~~12.1.6~~12.1.1.4 (1.1.2) Uranium Byproduct Cylinder (UBC) Storage Pad

~~(Approved per CC-LS-2009-0019) The UBC Storage Pad and UBC Basin are not will not be operational at the beginning of the IPO. Initial Plant Operations phase; however, these systems will be operational prior to the Production Phase 1. The UBC Storage Pad will be completed in sections. The first section will be completed prior to Production Phase 1. Section 12.1.2.9.4, Storage, discusses cylinder storage for IPO.~~

Because the UBC Storage Pad is not completed, accident sequences FF42-1, FF43-1, FF43-2, and FF44-1 and associated IROFS36c, 36e, 36f, and 36g are not applicable.

There is no accident sequence directly associated with the UBC Basin.

~~12.1.7~~12.1.1.5 (1.1.2) Central Utilities Building (CUB)

~~The CUB will not be is not operational as described in ISA Summary § 3.3.1.7. However, \$ systems required for Initial Plant Operation will be are ready in sufficient capacity to support~~

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12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

plant operations. The following list describes ~~e~~Systems within the CUB that ~~will not be~~ are ~~ready~~ required for to support IPO ~~Initial Plant Operations~~ are as follows:

12.1.1.5.1 Centrifuge Cooling Water

~~☐(Approved per CC-EG-2008-0392, Rev.0)~~The Centrifuge Cooling Water System (CCWS) ~~will be~~ is operational with the exception of the cooling towers. A bypass line has been installed to isolate the cooling towers and only the Centrifuge Water Heat Exchanger (cooled by CCW Chillers) is currently utilized as a heat removal source for the CCWS. This arrangement supports all operable cascades for IPO.

There is no accident sequence or IROFS directly associated with Centrifuge Cooling Water System.

- ~~• (CC-EG-2010-0008 Pending)~~The DI Water System ~~will be~~ is brought online as needed to support make-up water requirements after the initial system fill is made. A temporary skid-mounted polisher will be is installed until the permanent equipment is operational in the CUB.

12.1.1.5.2 Electrical Power Distribution

~~☐(Approved by CC-LS-2010-0001)~~Normal electrical power is supplied to the CRDB ~~will not be~~ is not available SBM1001 and the CAB. Howe ver, Depending on the scheduled completion date for storage area within the CRDB, alternate power is available and will may need to be supplied as necessary.

There is no accident sequence or IROFS directly associated with the electrical power distribution system. ~~(Approved by CC-LS-2010-0001)~~ Final commissioning and acceptance will be is in progress when Initial Plant Operations begin. These activities will be are complete in sufficient time to support continued plant operations.

12.1.1.6 ~~(1.1.2)~~ Administration Building

~~(Approved per CC-LS-2009-0002, Rev. 2)~~The Administration Building ~~will not be~~ is not operational. Until building completion, the staff will continue to be housed in temporary buildings on the east end of the facility. The Administration Building lobby is designed to act as an assembly area for emergency planning purposes. Alternate assembly areas are designated for assembly until completion for the Administration Building.

There is no accident sequence or IROFS directly associated with the Administration Building.

~~A.~~The Administration Building provides over 50 work locations for plant office staff. ~~Until building completion, the staff will continue to be housed in temporary buildings on the east end of the facility.~~

~~B.~~The Administration Building lobby is designed to act as an assembly area for emergency planning purposes. ~~Alternate assembly areas are designated for assembly until completion for the Administration Building.~~

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12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

23.0.5 Not Used

12.1.1.7 ~~(1.1.2)~~ Site Security Buildings

~~(Approved by CC-LS-2010-0001)~~ The main Security Building at the entrance of the facility will not be is operational for access to the Controlled Access Area (CAA). Instead, the existing security trailer will continue to be used. Vehicular traffic passes through additional security checkpoints before being allowed to park. Parking is located outside of the Controlled Access Area (CAA) security fence. Visitor passes are issued at a temporary security trailer located at the south east entrance to the facility.

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There is no accident sequence or IROFS directly associated with the Security Building

~~12.2.12.1.2~~ ~~(1.1.3)~~ Process Differences for Initial Plant Operation (IPO)

The differences between the processes as described in ISA Summary § 3.4 and Initial Plant Operation are as follows:

23.0.8 ~~(1.1.3.1)~~ Overview of Gas Centrifuge Enrichment Process

The overview of the gas centrifuge enrichment process is as described in ISA Summary § 3.4.1.

12.1.2.1 ~~(1.1.3.2)~~ UF₆ Feed System

~~(Approved per CC-LS-2009-0002, Rev. 1)~~ The UF₆ Feed and Feed Purification Systems will be is are operational as described in ISA Summary § 3.4.2 except a minimum of three (3) Solid Feed Stations (SFS) and one (1) Feed Purification Low temperature Take-off Station (LTTS) are required to be operable for FCOL enrichment operations. As IPO progresses, additional stations are completed and brought online as needed to support the incremental start up of cascades. The second Feed Purification Station (if operable) and all operations SFS not in use for enrichment operations contain a full feed cylinder at the beginning of IPO.

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Accident sequences UF1-1, UF2-1, and associated IROFS4 and 5 are applicable. the UF₆ Feed System will initially contain a sufficient number of operational UF₆ Solid Feed Stations to maintain operational flexibility for the operational Cascade Hall. These Feed Stations will be are brought into service as needed to support incremental startup of cascade modules. All operational feed stations will contain a full feed cylinder. Additional feed cylinders will be are stored in the spare tails stations to provide enough feed stock (and eventually tails storage) for approximately 3 months of operation before requiring additional storage space.

~~12.2.3~~ 12.1.2.2 ~~(1.1.3.2)~~ Cascade System

~~(Approved per CC-LS-2009-0002, Rev. 1)~~ The Cascade System will be is operational as described in ISA Summary § 3.4.3 with the exception that only one cascade module will be is on line operational at the beginning of Initial Plant Operation IPO. Cascades modules will be are brought online incrementally when the centrifuges within each cascade and all support equipment related to each cascade module are commissioned. At the end of IPO, cascade modules 1 through 6 are operable.

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Accident sequence EE-SEISMIC-SBM and associated IROFS41 is applicable.

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~~12.2.4~~12.1.2.3 (1.1.3.2) Product Take-off System

(Approved per CC-LS-2009-0002, Rev. 1) The Product Take-off System ~~will be~~ is operational as described in ISA Summary § 3.4.4 except a minimum of three (3) Product LTTS are required to be operable for FCOL enrichment operation. As IPO progresses, additional Product LTTS are brought online as needed to support the incremental start up of cascades. All operable Product LTTS not in use for enrichment operations contain an empty Product Cylinder at the beginning of IPO.

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~~Accident sequences PT2-1 and PT2-2 and associated IROFS1 and IROFS2 are applicable, with the following exception. The Product Low Temperature Takeoff Stations and supporting equipment may not all be in operation when the first cascade is started up. Each Product Low Temperature Takeoff Station will be~~ is brought online as needed to support the incremental start up of cascades.

~~12.2.5~~12.1.2.4 (1.1.3.2) Tails Take-off System

(Approved per CC-LS-2009-0002, Rev. 1) The Tails Take-off System ~~will be~~ is operational as described in ISA Summary § 3.4.5-5 except a minimum of three (3) Tails LTTS are required to be operable for FCOL enrichment operations. As IPO progresses, additional Tails LTTS are brought online as needed to support the incremental start up of cascades. All operational stations not in use for enrichment operations contain a full feed cylinder at the beginning of IPO, with the exception that all stations not in use will initially contain a full feed cylinder. Once an in-service feed cylinder is emptied, it will be is switched with a full feed cylinder from ~~the~~ tails station. The empty feed cylinder ~~can then be~~ is then used for normal tails take-off. This cylinder storage strategy will allow approximately 3 months of operation before additional cylinder storage space is required. ~~In addition, the Tails Low Temperature Takeoff Stations and supporting equipment may not all be in operation when the first cascade is started up. Each Tails Low Temperature Takeoff Station will be~~ is brought online as needed to support the incremental start up of cascades. ~~Sufficient Tails Stations will be~~ are available at all times to accommodate peak flow from the cascades.

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Accident sequence TT2-1 and associated IROFS1 and 2 are applicable.

~~12.2.6~~12.1.2.5 (1.1.3.2) Product Blending System

(Approved per CC-LS-2009-0002, Rev. 1) The Product Blending System ~~will not be~~ is not operational as described in ISA Summary § 3.4.6. ~~The Blending System and is not needed for Initial Plant Operations. It will be~~ is in operation when needed to support plant operations. IPO; however, the Blending Donor and Receiver Stations are operable for storage of full product cylinders.

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Accident sequences PB1-1 and PB2-1 and associated IROFS1,2,4, and 5 are applicable.

~~12.2.7~~12.1.2.6 (1.1.3.2) Product Liquid Sampling System

(Approved per CC-LS-2009-0002, Rev. 1) The Product Liquid Sampling System ~~will not be~~ is not operational at Initial Plant Operation and is not required for IPO. The Product Liquid Sampling Autoclaves will be are unavailable. Autoclaves will be are operational for Production Phase 1 to provide sampling capability for product that is ready for shipment.

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12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

Because the autoclaves are not available, accident sequences PB4-1, PB4-2, PB4-3, PB4-4, EE-TORNADO MISSILE-SBM PUBLIC, and EE-SEISMIC-SBM and associated IROFS10, 11, 12, and 28 are not applicable. (Note: the seismic and tornado events are applicable to the SBM but the autoclave contribution to the total release [and therefore IORFS28] is not applicable.)

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12.1.2.7 Contingency Dump System

~~(Approved per CC-LS-2009-0002, Rev. 1)~~ The Contingency Dump System will be is operational as described in ISA Summary § 3.4.8. Each operating cascade module has its own dedicated Contingency Dump System available for use. As additional cascades are completed, additional contingency dump components are installed and made operational in the process services corridor to support incremental plant start up and expansion.

There is no accident sequence or IROFS directly associated with the Contingency Dump System.

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12.2.012.1.2.8 Gaseous Effluent Vent Systems

~~(Approved by CC-EG-2009-0293) The Gaseous Effluent Ventilation System (GEVS) Room is not operational. The GEVS System is constructed as three separate systems, Pumped Extract GEVS, Local Extract GEVS, and Fume Hood GEVS (Fume Hood GEVS is not required for Initial Plant Operations). Pumped Extract GEVS is permanently installed in the UF₆ Handling Area of SBM 1001 and is operational for Initial Plant Operations. The Pumped Extract GEVS is temporarily connected to Local Extract ductwork in the SBM to support Initial Plant Operations.~~

~~(CC-EG-2009-0101 Pending) When the GEVS Room is complete, the permanent Local Extract GEVS will be installed along with the Fume Hood GEVS in that room. Once these GEVS systems are operational, the Pumped Extract GEVS' temporary connection to the Local Extract ductwork will be isolated. The Gaseous Effluent Ventilation System (GEVS) Room will not be operational. The GEVS System will be constructed as three separate systems, Pumped Extract GEVS, Local Extract GEVS, and Fume Hood GEVS (Fume Hood GEVS is not required for Initial Plant Operations). The Gaseous Effluent Ventilation System (GEVS) is constructed as two separate systems, Pumped Extract GEVS and CRDB GEVS. Pumped Extract GEVS is permanently installed in the UF₆ Handling Area of SBM 1001 and is operational for IPO. The local extract ductwork that is used in the SBM is temporarily connected to the Pumped Extract GEVS to support IPO.~~

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Because of this temporary cross-connection, there are limitations to the local extract capability. The following measures are in place to ensure adequate flow is provided at each local extract station:

- Only two local extract flexible hose stations are allowed to be open at any one time (IF the Ventilated Storage Room is online, THEN only one flexible hose station is allowed to be in use).
- Configuration control is maintained by the Shift Manager and the use of caution tags on the local extract flexible hose station isolation valves.

All GEVS accident sequences (CL3-1, CL3-2, CL3-3, VR1-1, VR1-2, VR 2-2, and FF25-2) and associated IROFS (IROFS20, 21, 24a, 24b, and 37) are for CRDB operations and therefore not applicable to IPO.

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Accident sequence LOSS OF SAFE-BY-DESIGN ATTRIBUTE is applicable for the pumped Extract GEVS.

There is no accident sequence or IROFS directly associated with the local extract function of the CRDB GEVS.

A. Pumped Extract GEVS will be permanently installed in the UF₆ Handling Area of SBM 1001 and will be operational for Initial Plant Operations. The Pumped Extract GEVS will be temporarily connected to Local Extract ductwork in the SBM to support Initial Plant Operations.

B. When the GEVS Room in the CRDB is complete, the permanent Local Extract GEVS will be installed along with the Fume Hood GEVS in that room. Once these GEVS systems are operational in the CRDB, the Pumped Extract GEVS' temporary connection to the Local Extract ductwork will be isolated.

12.2.10 Centrifuge Test and Centrifuge Post Mortem Processes

The Centrifuge Test and Centrifuge Post Mortem Facility will be operational as described in ISA Summary § 3.4.10.

12.2.11 12.1.2.9 (CC-OP-2009-0002 Pending) Material Handling Processes

12.1.2.9.1 Cylinder Receipt and Shipping

During initial plant operations IPO, cylinders are shipped and received via a Vehicle Loading and Unloading Area loading platform on the West side of the UF₆ Handling Area of SBM-1001 (West side SBM 1001). The Vehicle Loading and Unloading Area West side SBM 1001 space for the following services:

- Cylinder loading and unloading
- Inventory weighing
- Secure internal storage
- Preparation and storage area for overpack/protective structural packaging.

The cylinders are received, shipped offsite, stored, and transferred to and from the UF₆ Handling Area at the Vehicle Loading and Unloading Area until the CRDB and UBC Storage Pad becomes operational.

Full feed cylinders, empty feed cylinders, semi finished product cylinders, and UBCs are stored in the UF₆ Handling Area until the CRDB and UBC Storage Pad become operational.

12.1.2.9.2 Description

Commercial transport tractors are disconnected from the trailers carry containers and connected to LES yard tractors which comply IROFS36c (diesel fuel capacity less than 280 L (74 gal)). The yard tractor will delivers UF₆ cylinders (i.e., full 48Y feed cylinders, and new or cleaned 30B product cylinders) to the and Unloading Area on the west side, south end of SBM1001 West

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side SBM-1001 receipt platform. Cylinders are unloaded with a gantry crane of sufficient capacity to perform all required cylinder movements. The gantry crane will lift and transfer the cylinder to the rail transporter that sits on rails that are extended outside the SBM into the Vehicle Loading and Unloading Area. On completion of receipt inspection, the rail transporter will move the cylinder inside the UF₆ Handling Area. Cylinders are removed from the facility in the same fashion.

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12.1.2.9.3 Equipment

The following equipment is used for cylinder handling on the West side SBM-1001 receipt platform.

A. Vehicle Loading and Unloading Area Platform

The Vehicle Loading and Unloading Area vehicle loading and unloading platform is located adjacent to the West side SBM-1001 equipment hatch. This platform provides a safe method of transfer from the vehicle trailer to rail transporter located on the platform.

Accident sequence FF7-1 and associated IROFS36c is applicable to the LES yard tractor at the Vehicle Loading and Unloading Area.

B. Gantry Crane:

A dedicated gantry crane is used to handle cylinders on the vehicle loading and unloading area platform. The crane spans the width of the loading platform to access vehicle trailers and the rail transporter. The hoist has a maximum lift of approximately 6.1 m (20 ft). Crane specifications are as follows:

•	Span	11.3 m (37 ft)
•	Capacity	20 MT (44,100 lb)
•	Hoist lift height	3.1 m (20 ft)
•	Hoist lift speed	3 m/min & 0.5 m/min (10 ft/min & 1.6 ft/min)
•	Travel length	7.9 m (26 ft)
•	Bridge travel speed (VFD)	19.8 m/min (65 ft/min)
•	Brake type	Direct Current Disk

C. Scale:

(Approved by CC-EG-2009-0294) Inventory Weighing is performed using a temporary scale in the UF₆ Handling Area of SBM-1001. The scale is identical to the scales described in ISA Summary § 3.4.11.1.2. C. Each cylinder that enters or exits the UF₆ Handling Area during the Initial Operations Phase IPO is weighed. A weigh scale capable of weighing a load of 17 MT

12.1 FACILITY DIFFERENCES FOR INITIAL PLANT OPERATIONS (IPO)

(37,500 lb) and capable of accepting a load of 20 MT (44,100 lb) is installed. The scale is capable of weighing to a tolerance of ± 2.5 kg (± 5.5 lb). The scale has reader and printout facilities.

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There is no accident sequence of IROFS directly associated with the weigh scales.

D. Powered Vehicles And Rail Transporters:

LES yard tractors that comply IROFS36c (diesel fuel capacity less than 280 L) are utilized to deliver the vehicle trailer containing cylinders to the Vehicle Loading and Unloading Area West side SBM 1001 receipt platform. The gantry crane will lift and transfers the cylinder to the rail transporter that sits on rails extended outside the SBM into the Vehicle Loading and Unloading Area stillage just outside the SBM accessible to the rail transporter. On completion of receipt inspection, the rail transporter will retrieve the cylinder for use. Cylinders are removed from the facility in the same fashion.

Accident sequence FF7-1 and associated IROFS36c is applicable to the LES yard tractors at the Vehicle Loading and Unloading Area.

There is no accident sequence or IROFS directly associated with the Rail Transporter.

Cylinder Specifications:

As specified in § 3.4.11.1.3.

12.1.2.9.4 Storage

Storage is made available in phases. Initially, cylinders are stored in their respective stations in the UF₆ Handling Area. When available, the CRDB and the UBC Storage Pad is utilized for storage of cylinders.

During initial plant operations, cylinders are placed on and removed from delivery trucks using a gantry crane. They are moved inside the UF₆ area using the rail transporter and in the CRDB (when available in Production Phase 1) using the West Bridge Crane. The other bridge cranes in the CRDB are installed at a later date.

The UBC Storage Pad is not operational at Initial Plant Operations. It is completed in time to provide storage of cylinders while the CRDB is being finished.

A. For IPO, all operable feed, feed purification, product, tails, and blending stations contain a cylinder. All feed stations contain full feed cylinders. One feed purification LTTS contains an empty cylinder for purification operations. The other feed purification LTTS contains a full feed cylinder. Two Tails LTTS will contain empty cylinder to collect tails. The remaining operable Tails LTTS will contain full feed cylinders. All Product LTTS and the Blending System Donor Station and LTTS will contain empty Product Cylinders. For FCOI, one feed, product, and tails station will be in service and one in standby to support normal enrichment operations. When the first product cylinder is filled, the process will shift to the standby product station. This is a normal operation. However, the first two full product cylinders will be removed from the Product LTTS and replaced with empty product cylinders from the Blending System Donor Station or LTTS. The full product cylinder will be stored in the empty station that previously contained the empty product cylinder. As the remaining

Product cylinders are filled, they will remain in their respective Product LTTS for storage. This switching process is also used for feed and tails cylinders. As the feed cylinder empties, it will shift to the standby feed station. This is also a normal operation. The empty feed cylinder is then replaced with a full feed cylinder from a Tails LTTS. The empty feed cylinder is installed into the empty Tails LTTS and will eventually be used to collect tails. This switching of cylinders will allow approximately 3 months of operation before additional storage space is required.

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- B. In the event that additional storage is required, filled tails cylinders may be shipped off-site to a licensed facility until such time as the UBC Storage Pad is operational. In addition, filled product cylinders may be shipped off-site to a licensed storage facility until such time as the site homogenization, sub-sampling, and analysis capabilities are established.

Accident sequences UF1-1, UF2-1, PT2-1, PT2-2, TT2-1, PB1-1, PB2-1, PB2-2, and CP1-2 and associated IROFS1, 2, 4, 5, and 16a are applicable.

Cylinder Deliveries

As specified in § 3.4.11.1.7, with the exception that the numbers of deliveries and shipments are less during the Initial Plant Operations Phase due to limited initial production and storage capacity.

~~C. During initial plant operations, cylinders will be shipped and received via a loading platform on the West side of the UF₆ Handling Area of SBM 1001. Inventory Weighing will be performed using temporary scales in the UF₆ Handling Area of SBM 1001. Storage will be made available in phases. Initially, cylinders will be stored in their respective stations. When available, the South end of the CRDB and the UBC Storage Pad will be utilized for storage of cylinders.~~

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~~D. During initial plant operations, cylinders will be placed on and removed from delivery trucks using a mobile crane of sufficient capacity. They will be moved inside the UF₆ area using the rail transporter and in the CRDB (when available in Production Phase 1) using the West Bridge Crane. The other bridge cranes in the CRDB will be installed at a later date.~~

~~E. The UBC Storage Pad will not be operational at Initial Plant Operations. It will be completed in phases in time to provide storage of cylinders while the CRDB is being finished.~~

12.1.3 Utility and Support System Differences for Initial Plant Operations (IPO)

~~The differences between the utility and support systems as described in § 4.9, 6.2, and Initial Plant Operation are as follows:~~

~~A. The Building Ventilation will be is operational as described in ISA Summary § 3.5.1.~~

~~B. The Electrical System will be is operational as described in ISA Summary § 3.5.2.~~

~~C. The Compressed Air System will be is operational as described in ISA Summary § 3.5.3.~~

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~~D. Not Used (Approved per CC ECLS 201000-00028, Rev. 1 Pending) The Deionized Water System will not be operational. Initial system fill and makeup water, if required, will be performed by an external source, such as a tanker truck. A temporary skid-mounted polisher will be installed until the permanent equipment is operational in the CUB.~~

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12.1.3.1 Centrifuge Cooling Water

~~(6.2.1.2.8) (Approved per CC EG 2008-0392, Rev. 0) The Centrifuge Cooling Water (CCW) System will be is operational with the exception of the cooling water towers. The cooling water towers will be are bypassed and heat removal will be is performed by the CCW heat exchanger cooled by the CCW chiller units. This arrangement can supports all operable cascades for IPO several cascades on line [CC EG 2008-0392]. When the cooling towers are completed and additional cooling is needed, the bypass valve will be is closed and normal operation will commence.~~

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There is no accident sequence directly associated with CCWS.

~~D. The Sewage System will be is operational as described in ISA Summary § 3.5.6.~~

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~~E. The Communication and Alarm Annunciation System will be is operational as described in ISA Summary § 3.5.7.~~

F. Not Used

~~G. The Control System will be is operational as described in ISA Summary § 3.5.9.~~

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~~H. The Standby Diesel Generator System will be is operational as described in ISA Summary § 3.5.10.~~

~~I. (6.2.1.2.4) The Nitrogen System will be is operational as described in ISA Summary § 3.5.11.~~

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12.1.3.2 Reserved

~~L. (Approved per CC EG 2010-0005 Rev. 1) The Liquid Effluent Collection and Treatment System (LECTS) will not be is not operational. Instead, the various types of aqueous and non-aqueous liquid wastes generated by plant operations and processes in the facility will be are collected and either shipped off site to an appropriate treatment and disposal facility or stored on site in accordance with appropriate radiological and criticality safety controls until the Liquid Effluent Collection and Treatment Facility is completed.~~

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12.1.3.3 Solid Waste Collection System

~~M. The Solid Waste Collection System will not be is not operational. Solid wastes will either be stored on site using appropriate chemical, radiological, and criticality safety controls until the Solid Waste Collection Room is completed or shipped off site to a processing facility for treatment and/or disposal at a licensed facility.~~

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Because the Solid Waste Collection Room is not completed, accident sequences SW1-1 and SW1-2 and associated IROFS14a and IROFS14b are not applicable.

12.1.3.4 Decontamination Workshop

~~(4.9.1) (Approved per CC-LS-2009-0002, Rev. 1) The Decontamination Workshop will not be is not operational. Contaminated equipment will be is stored in accordance with appropriate chemical, radiological, and criticality safety controls until the Decontamination Workshop is completed or shipped off site to a processing facility for treatment and/or disposal at a licensed facility.~~

Because the Decontamination Workshop is not completed, accident sequence LOSS OF SAFE-BY-DESIGN ATTRIBUTE is not applicable.

12.1.3.5 PFPE Oil Recovery System

~~O. (Approved per CC-LS-2009-0002, Rev. 2) The PFPE Oil Recovery System will not be is not operational; however, the system has no impact on any safety aspect of facility operation. PFPE oil will either be appropriately stored on site until the system is operational or disposed of at a certified disposal facility.~~

P. Not Used

12.1.3.6 Ventilated Room

~~Q. (Approved per CC-EG-2009-0369 Pending) The Ventilated Room will not be is not operational. A Ventilated Storage Room temporary room will be is has been constructed in the UF₆ Handling Area in SBM-1001 to for limited storage of any equipment or waste that would normally be stored in the Ventilated Room during IPO initial Plant Operations as necessary. This room will be is connected to the Pumped Extract GEVS. The room will be is used for storage only; no processing of equipment or materials will be are conducted. Although a leaking valve on a cylinder containing UF₆ is not expected, if one is identified, the potential leakage will be is stopped in one of three ways depending on the nature of the damage. The valve will be is capped, the valve stem will be is tightened or the packing gland will be is tightened and the cylinder stored in an appropriate (feed or product) station until repairs can be conducted or the cylinder can be returned to the vendor.~~

Transitional accident sequences TVR1-1, TVR1-2, and TVR1-3 associated with this room have been identified that require implementation of existing IROFS14a and 14b, and IROFS31a, 31b, and 31c. See ISA Summary Tables 4-4, Transitional Accident Sequence and Risk Index, and 4-5, Transitional Accident Sequence Descriptions.

12.1.3.7 Chemical Laboratory

~~R. (Approved per CC-LS-2009-0002, Rev. 1) The Chemistry Laboratory will not be is not operational. Instead, samples will be is collected and shipped to a certified testing facility for analysis. The sample containers will not be are not returned to LES, but will be are disposed of by the facility.~~

Because the Chemical Laboratory is not completed, accident sequences CL3-1, CL3-2, and CL3-3 and associated IROFS24b, 43, and 46 and LOSS OF SAFE-BY-DESIGN ATTRIBUTE are not applicable.

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12.1.4 Safety Significance

Section 12.0 of the Safety Analysis Report has been initially established as an administrative change to describe the Phased Operation concept. There is no safety significance because none of the identified changes will be finalized and implemented until reviewed and approved in accordance with the LES configuration management program as described in § 11.1, Management Measures. Pursuant to 10 CFR 70.72, LES has established a system to evaluate, implement, and track each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel. Configuration management of IROFS, and any items that may affect the function of IROFS, is applied to all items identified within the scope of the IROFS boundary. All changes to structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary are evaluated before the change is implemented. If the change requires an amendment to the License, Nuclear Regulatory Commission approval is received prior to implementation.

All proposed changes described in Section 12.0 are tracked and evaluated per the LES configuration management program prior to implementation. As the changes are processed, Section 12.0 will be revised to incorporate changes to the facility, processes, and programs. Section 12.0 documents all site changes facilitated as a result of the Phased Operation approach.

12.2 PRODUCTION PHASE 1

Several other support functions are available and ready to support plant operations. This provides additional cylinder storage, the ability to sample product for shipment to customers, and other chemistry activities.

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12.3 Production Phase 2

12.3 PRODUCTION PHASE 2

Functions supporting sample analysis, wet and dry low level waste collection and treatment, and radioactive decontamination and maintenance of plant equipment are available. Additional cascades and support equipment are added to increase production, but the plant is fully capable of carrying out continuous commercial production.

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12.4 Production Phase 3

12.4 PRODUCTION PHASE 3

Cascade modules in Cascade Hall 1002 are started up incrementally as needed to support continued plant expansion. This incremental start up continues until Cascade Halls 1001 and 1002 are fully operational.

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12.5 Production Phase 4

12.5 PRODUCTION PHASE 4

Construction of remaining SBMs is completed and cascade modules started up incrementally as needed to support final plant expansion. This incremental start up continues until all Cascade Halls are fully operational. The duration of this phase is dependent on the final approved design and SWU capacity.

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