Enclosure 3
Clean copy of Safety Analysis Report,
Revision 31

SAFETY ANALYSIS REPORT
Revision 31

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Summary of Changes for Revision 30		
Issue / Date	**Change**	**Description of Change**
30a 02-25-11	LBDCR-11-0007 02-01-11	Revisions based on condition reports CC-EG-2010-0336; 70.72 = 2010-0060
	LBDCR-11-0012 02-11-11	Contingency storage of product cylinders in the UF_6 Handling area during initial plant operation. CC-OP-2010-0004; 70.72 = 2011-0083
30b 04-05-11	LBDCR-11-0017 03-23-11	Remove “SA, 2001” after MONK 8A throughout. CC-EG-2011-0088; 70.72 = 2011-0190
30c 05-10-11	LBDCR-11-0014 04-05-11	Clarification to describe the difference between the 10,000 year earthquake and the NEF DBE CC-EG-2011-0007; 70.72 = 2011-0143
	LBDCR-11-0019 04-06-11	Incorporation of various corrective actions from condition reports CC-EG-2011-0027; 70.72 = 2011-0216
	LBDCR-11-0016 04-12-11	Update figures to the as built drawings CC-EG-2011-0033; 70.72 = 2011-0231
	LBDCR-11-0020 05-06-11	Changes the ownership information to be consistence with current ownership information. CC-LS-2011-001; 70.72 = 2011-0224
30d 06-13-11	LBDCR-11-0025 05-23-11	Remove SPCC references CC-EN-2011-0003; 70.72 = 2011-0286
	LBDCR-11-0027 05-23-11	Removal of Ventilated Storage Room from UF_6 Handling Area in SBM-1001 CC-OP-p2011-0002 rev 1; 70.72 = 2011-0290
	LBDCR-11-0029 05-26-11	Removed accident sequence PB1-3 & changes IROFS45 to only apply to the CRDB. CC-OP-2011-0010; 70.72 = 2011-0295

Summary of Changes for Revision 30		
Issue / Date	**Change**	**Description of Change**
	LBDCR-11-0030 05-27-11	Replace specific listings of the current enrichment limit of 5% with references to the license limit for isotope U-235 CC-OP-2011-0003; 70.72 = 2011-0314
30d Continued	LBDCR-11-0031 06-03-11	Organizational changes (VP of Compliance/GC to VP of Regulatory Affairs; Director of Compliance; Director of Plant Support responsibilities; Training Manager) CC-LS-2011-0002; 70.72 = 2011-0308
30e 08/08/11	LBDCR-11-0043 7-1-2011	Historical note is being added to SAR Section 3.4.22 LAR 11-04
	LBDCR-11-0034 6-30-11	Approved LAR 08-07 revised the SAR and QAPD so that Structures, Systems, and Components that are not essential to IROFS yet can affect and IROFS are not QL-1 CC-LS-2010-0016, r1 70.72= 2011-0410
	LBDCR-11-0038 7-13-11	LBDs used to either define Phased Operation or reference the definition of the Phased Operation CC-OP-2011-0008; 70.72 = 2011-0368
30f 08/17/11	LBDCR-11-0035 8-5-11	Implement Procurement of certain QL-1F items where it became clear that certain information contained in the NRC SER for LAR 10-08 for implementation of Fire Protection Items Relied on For Safety CC-QA-2011-0001 Rev 2; 70.72 = 2011-0239
	LBDCR-11-0046 8-9-11	Allow the use of all safety analysis methods described in NUREG-1513 and revises the definition of "not credible" CC-OP-2011-0006; 70.72 = 2011-0441
30g 09/01/11	LBDCE-11-0028 5-19-11	The temporary operation of the Pump Extract GEVS and Local Extract GEVS in a cross-tied configuration until the completion of the CRDB and the commissioning of the Local Etract GEVS fan filter units CC-EG-2011-0015; 70.72 = 2011-0294

Summary of Changes for Revision 30		
Issue / Date	**Change**	**Description of Change**
31 10/13/11	LBDCR-11-0051 10-05-11	Addition of a storage location for sample containing UF_6. CC-OP-2011-0013; 70.72 = 2011-0545
	N/A	Submittal to NRC for non substantial changes previously approved by LES

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ACRONYMS AND ABBREVIATIONS

AC	alternating current
ACI	American Concrete Institute
ADEM	Alabama Department of Environmental Management
AEA	Atomic Energy Act
AEP	American Electric Power
AEGL	Acute Exposure Guideline Level
AHU	air handling unit
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ALI	Annual Limit on Intake
ANPR	Advance Notice of Proposed Rulemaking
ANS	American Nuclear Society
ANSI	American National Standards Institute
AP	air particulate
APE	area of potential effects
AQB	Air Quality Bureau
ASCE	American Society of Civil Engineers
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
ASNT	American Society of Nondestructive Testing
ASTM	American Society for Testing Materials
ATSDR	Agency for Toxic Substances and Disease Registry
AVLIS	Atomic Vapor Laser Isotope Separation
BDC	baseline design criteria
BEA	Bureau of Economic Analysis
BLM	Bureau of Land Management
BMP	Best Management Practices
BNFL	British Nuclear Fuels
BNFL-EL	British Nuclear Fuels – Enrichment Limited
BOD	biochemical oxygen demand
BS	Bachelor of Science
CA	Controlled Area
CAA	Clean Air Act
CAAS	Criticality Accident Alarm System
CAB	Centrifuge Assembly Building

ACRONYMS AND ABBREVIATIONS

CAM	Continuous Air Monitor
CAP	Corrective Action Program
CBG	Census Block Group
CEDE	Committed Effective Dose Equivalent
CEQ	Council on Environmental Quality
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFO	Chief Financial Officer
CFR	Code of Federal Regulations
CHP	certified health physicist
CIS	Commonwealth of Independent States
CM	configuration management
COD	chemical oxygen demand
CRDB	Cylinder Receipt and Dispatch Building
CUB	Central Utilities Building
CVRF	Central Volume Reduction Facility
CWA	Clean Water Act
D&D	decontamination and decommissioning
DAC	derived air concentration
DBA	design basis accident
DBE	design basis earthquake
DCF	dose conversion factor
DE	Dose Equivalent
DEIS	Draft Environmental Impact Statement
DI	deionized
DOC	United States Department of Commerce
DOE	United States Department of Energy
DOI	United States Department of Interior
DOT	United States Department of Transportation
E	east
EDE	Effective Dose Equivalent
EECP	Entry/Exit Control Point
EIA	Energy Information Administration
EIS	Environmental Impact Statement
EJ	Environmental Justice
EMS	Emergency Medical Services

ACRONYMS AND ABBREVIATIONS

EOC	Emergency Operations Center
EPA	United States Environmental Protection Agency
EPCRA	Emergency Planning and Community Right-to-Know Act
EPRI	Electric Power Research Institute
eqs.	equations
ER	Environmental Report
ERPG	Emergency Response Planning Guideline
ENE	east north east
ESE	east south east
ETTP	East Tennessee Technology Park
FEIS	Final Environmental Impact Statement
FEMA	Federal Emergency Management Agency
FHA	fire hazards analysis
FNMC	Fundamental Nuclear Material Control
FR	Federal Register
FWPCA	Federal Water Pollution Control Act
GDP	Gaseous Diffusion Plant
GET	General Employee Training
GEVS	Gaseous Effluent Vent System
GPS	Global Positioning System
HEPA	high efficiency particulate air
HEU	highly enriched uranium
HMTA	Hazardous Materials Transportation Act
HS&E	Health, Safety, and Environment
HUD	United States Department of Housing and Urban Development
HVAC	heating, ventilating, and air conditioning
HWA	Hazardous Waste Act
HWB	Hazardous Waste Bureau
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INFL	International Nuclear Fuels Plc
I/O or I-O	input/output
IPD	Implicit Price Deflator
IROFS	items relied on for safety
ISA	Integrated Safety Analysis

ACRONYMS AND ABBREVIATIONS

ISO	International Organization for Standardization
JCIDA	Jackson County Industrial Development Authority
LAN	local area network
LCC	local control center
LCD	local climatic data
L_{dn}	Day-Night Average Sound Level
L_{eq}	Equivalent Sound Level
LES	Louisiana Energy Services
LEU	low enriched uranium
LLC	Limited Liability Company
LLD	lower limits of detection
LLNL	Lawrence Livermore National Laboratory
LLW	low-level waste
LOI	local operator interface
LQ	Location Quotients
LTA	lost time accident
LTC	load tap changer
LTTS	Low Temperature Take-off Station
M&TE	measuring and test equipment
MAPEP	Mixed Analyte Performance Evaluation Program
max.	maximum
MC&A	material control and accountability
MCL	maximum contaminant level
MCNP	Monte Carlo N-Particle
MDA	minimum detectable activity
MDC	minimum detectable concentration
ME&I	mechanical, electrical and instrumentation
min.	minimum
MM	modified mercalli
MMI	modified mercalli intensity
MOU	Memorandum of Understanding
MOX	mixed oxide fuel
MUA	multi-attribute utility analysis
N	north
NAAQS	National Ambient Air Quality Standards

ACRONYMS AND ABBREVIATIONS

NASA	National Aeronautic Space Administration
NCA	Noise Control Act
NCRP	National Council on Radiological Protection and Measurements
NCS	nuclear criticality safety
NCSE	nuclear criticality safety evaluation
NDA	Non-destructive assessment
NE	Northeast
NEF	National Enrichment Facility
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NESHAPS	National Emission Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NHPA	National Historic Preservation Act
NELAC	National Environmental Laboratory Accreditation Conference
NIOSH	National Institute of Occupational Safety and Health
NIST	National Institute of Standards and Technology
NM	New Mexico
NMAC	New Mexico Administrative Code
NMDGF	New Mexico Department of Game and Fish
NMED	New Mexico Environmental Department
NMHWB	New Mexico Hazardous Waste Bureau
NMRPR	New Mexico Radiation Protection Regulations
NMSA	New Mexico State Agency
NMSE	New Mexico State Engineer
NMSHPO	New Mexico State Historic Preservation Office
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

ACRONYMS AND ABBREVIATIONS

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	piping and instrumentation diagrams
p.	page
PA	public address
PCM	Personnel Contamination Monitor
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
PIA	Potentially Impacted Area
PLC	Programmable Logic Controllers
PM	preventive maintenance
$PM_{2.5}$	particulates $\leq 2.5\mu m$
PM_{10}	particulates $\leq 10\mu m$
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

ACRONYMS AND ABBREVIATIONS

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
RCA	Radiologically Controlled Area
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
SBM	Separations Building Module
Sc.D.	Doctor of Science
SCRAM	Support Center for Regulatory Air Models
SDWA	Safe Drinking Water Act
SE	southeast
SER	Safety Evaluation Report
SHPO	State Historic Preservation Officer
SILEX	Separation of Isotopes by Laser Excitation
SNM	special nuclear material
SPCC	spill prevention, control, and countermeasures
SPL	Sound Level Pressure
SRC	Safety Review Committee
SSC	structure, system, and component
SSE	safe shutdown earthquake
SSE	south-southeast
SSW	south-southwest
STEL	short term exposure limits
STP	standard temperature and pressure

ACRONYMS AND ABBREVIATIONS

SVOC	semivolatile organic compounds
SW	southwest
SWPPP	Storm Water Pollution Prevention Plan
TDEC	Tennessee Department of Environment and Conservation
TDS	Total Dissolved Solids
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TN	Tennessee
TSB	Technical Services Building
TSP	total suspended particulates
TVA	Tennessee Valley Authority
TWA	time weighted average
TWDB	Texas Water Development Board
TX	Texas
UBC	Uranium byproduct cylinder
UCL	Urenco Capenhurst Limited
UCN	Ultra-Centrifuge Netherlands NV
UNAMAP	Users Network for Applied Modeling of Air Pollution
UPS	uninterruptible power supply
US	United States
USACE	United States Army Corps of Engineers
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USDA	United States Department of Agriculture
USFWS	United States Fish and Wildlife Service
USGS	United States Geological Survey
UV	ultraviolet
VOC	volatile organic compound
W	West
WCS	Waste Control Specialists
WIPP	Waste Isolation Pilot Plant
WMA	wildlife management area
WNA	World Nuclear Association
WNW	west-northwest
WQB	Water Quality Bureau
WQCC	Water Quality Control Commission

ACRONYMS AND ABBREVIATIONS

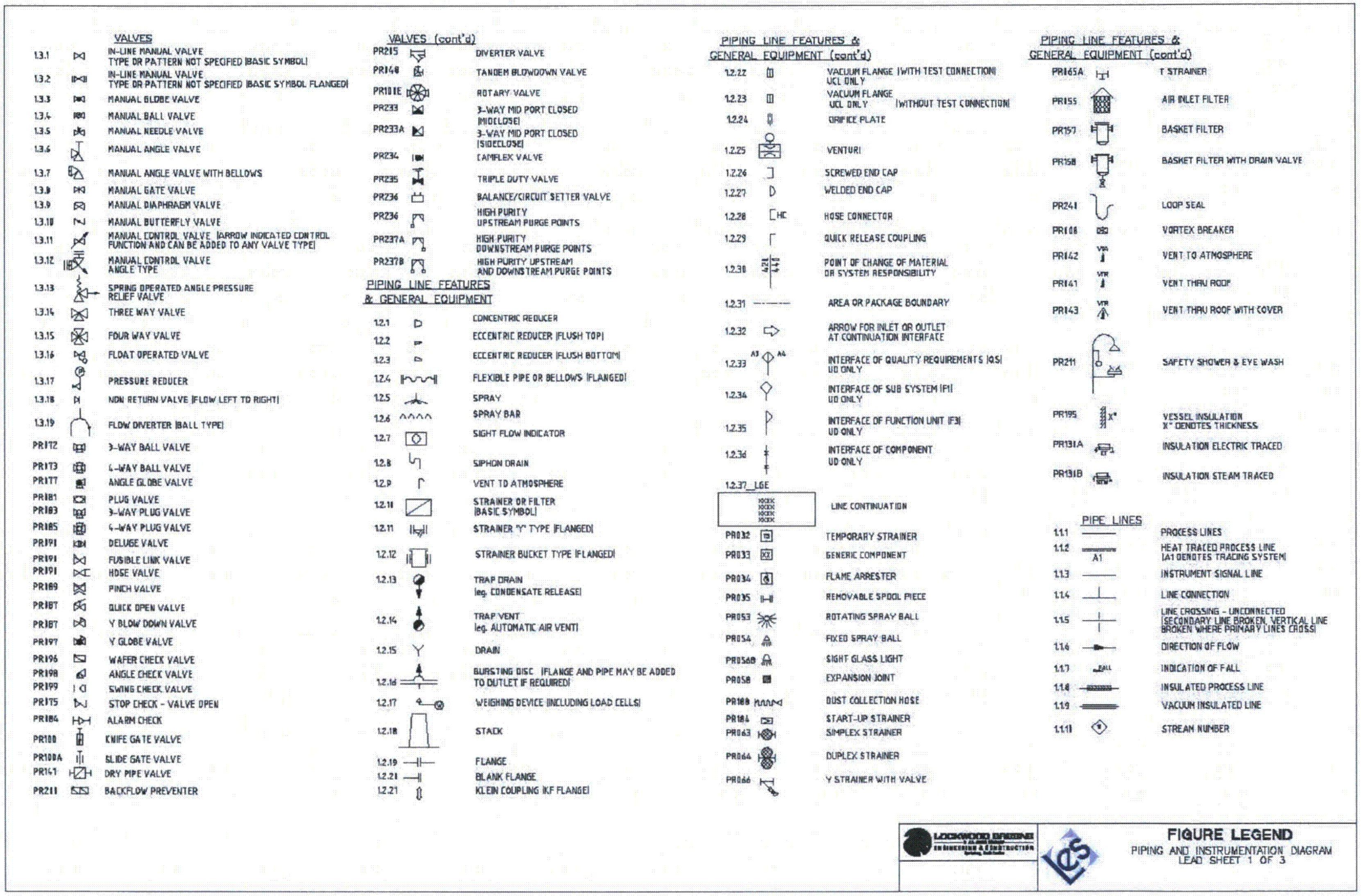
WSW	west-southwest

UNITS OF MEASURE

Bq	Becquerel
BTU	British thermal unit
°C	degrees Celsius
Ci	curie
cm	centimeter
d	day
dB	decibel
dBA	decibel A-weighted
dpm	disintegrations per minute
°F	degrees Fahrenheit
ft	feet
g	gram
g_a	gravitational acceleration
gal	gallon
gpm	gallons per minute
Gy	Gray
ha	hectares
hp	horsepower
hr	hour
Hz	hertz (cycle per second)
in	inch
in. H_2O	inches of water (column)
J	Joule
kg	kilogram
km	kilometer
kWh	kilowatt-hour
L	liter
lb	pound
lbs	pounds
m	meter
mbar abs	millibar absolute
mbarg	millibar gauge
MBq	megabecquerel
mi	mile
min	minute

UNITS OF MEASURE

M_N	local magnitude
Mo	month
msl	mean sea level
MT or t	metric ton
MTU	Metric ton uranium
oz	ounce
Pa	pascal
ppb	parts per billion
ppm	parts per million
psia	pounds per square inch absolute
psig	pounds per square inch gauge
R	Roentgen
rad	radiation absorbed dose
rem	Roentgen equivalent man
scfm	standard cubic feet per minute
s	second
Sv	sievert
SWU	separative work unit
µmhos	micromhos
V	volt
VA	volt-ampere
W	watt
$^w/_o$	weight percent
χ/Q	atmospheric concentration per unit source
yd	yard
yr	year
σ	standard deviation
Pico (p)	X 10^{-12}
Nano (n)	X 10^{-9}
Micro (µ)	X 10^{-6}
Milli (m)	X 10^{-3}
Centi (c)	X 10^{-2}
Kilo (k)	X 10^{3}
Mega (M)	X 10^{6}



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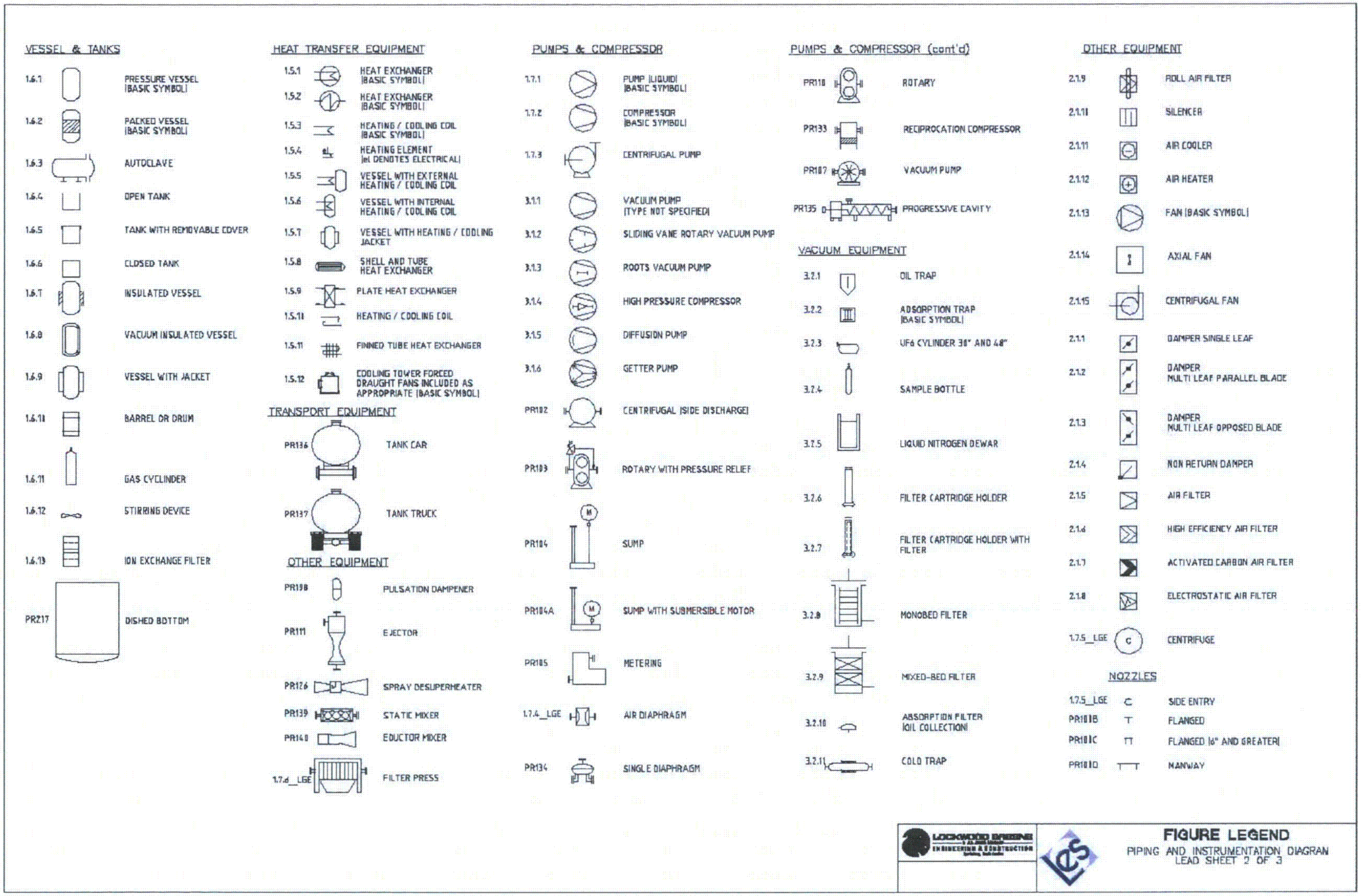

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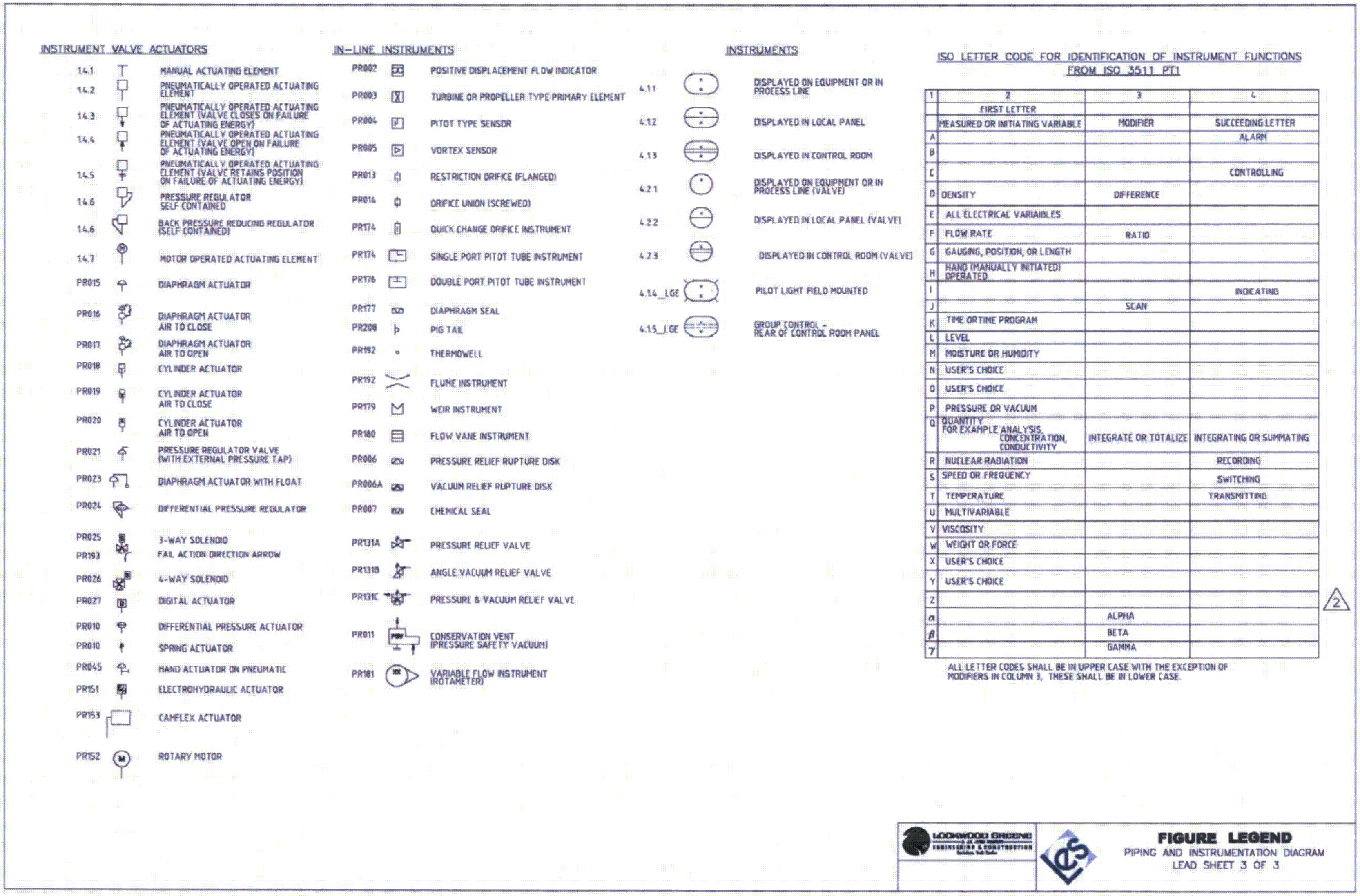

1	2	3	4
	FIRST LETTER		
	MEASURED OR INITIATING VARIABLE	MODIFIER	SUCCEEDING LETTER
A			ALARM
B			
C			CONTROLLING
D	DENSITY	DIFFERENCE	
E	ALL ELECTRICAL VARIAIBLES		
F	FLOW RATE	RATIO	
G	GAUGING, POSITION, OR LENGTH		
H	HAND (MANUALLY INITIATED) OPERATED		
I			INDICATING
J		SCAN	
K	TIME OR TIME PROGRAM		
L	LEVEL		
M	MOISTURE OR HUMIDITY		
N	USER'S CHOICE		
O	USER'S CHOICE		
P	PRESSURE OR VACUUM		
Q	QUANTITY FOR EXAMPLE ANALYSIS, CONCENTRATION, CONDUCTIVITY	INTEGRATE OR TOTALIZE	INTEGRATING OR SUMMATING
R	NUCLEAR RADIATION		RECORDING
S	SPEED OR FREQUENCY		SWITCHING
T	TEMPERATURE		TRANSMITTING
U	MULTIVARIABLE		
V	VISCOSITY		
W	WEIGHT OR FORCE		
X	USER'S CHOICE		
Y	USER'S CHOICE		
Z			
α		ALPHA	
β		BETA	
γ		GAMMA	

Figure Legend Sheet 3 of 3

1.0 General Information

This section contains a general description and purpose of the Louisiana Energy Services (LES) National Enrichment Facility (NEF). The facility enriches uranium for producing nuclear fuel for use in commercial power plants. This Safety Analysis Report (SAR) follows the format recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility. The level of detail provided in this chapter is appropriate for general familiarization and understanding of the facility and processes. The information is to be used as background for the more detailed descriptions provided in other chapters of the license application. Cross-references to the more detailed descriptions are provided in this chapter. This chapter also provides information on the corporate structure and economic qualifications of LES.

t is not practical to refer to a specific edition of each code, standard, NRC document, etc throughout the text of this document. Instead, the approved edition of each reference that is applicable to the design, construction, or operation of the NEF is listed in ISAS Table 3.0-1.

1.1 Facility and Process Description

The NEF, a state-of-the-art process plant, is located in southeastern New Mexico in Lea County approximately 0.8 km (0.5 mi) west of the Texas state border. This location is approximately 8 km (5 mi) due east of Eunice and 32 km (20 mi) south of Hobbs.

The geographic location of the facility is shown on Figures 1.1-1, State Map, and 1.1-2, County Map.

This uranium enrichment plant is based on a highly reliable gas centrifuge process. The plant is designed to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream - enriched in the uranium-235 (^{235}U) isotope and a tails stream - depleted in the ^{235}U isotope. The process, entirely physical in nature, takes advantage of the tendency of materials of differing density to segregate in the force field produced by a centrifuge. The chemical form of the working material of the plant, uranium hexafluoride (UF_6), does not require chemical transformations at any stage of the process. This process enriches natural UF_6, containing approximately 0.711% ^{235}U to a UF_6 product, containing ^{235}U enriched up to 5 $^{w}/_{o}$.

The nominal capacity of the facility is 3 million separative work units (SWU) per year. The maximum gross output of the facility is slightly greater than 3 million SWU thus allowing for a production margin for centrifuge failures and occasional production losses during the operational lifetime of the facility.

Feed is received at the plant in specially designed cylinders containing up to 12.7 MT (14 tons) of UF_6. The cylinders are inspected and weighed in the Cylinder Receipt and Dispatch Building (CRDB) or UBC Storage Pad and transferred to the Separations Building Modules (SBMs). SBMs are divided into two Cascade Halls, and each Cascade Hall is comprised of multiple cascades. Each Cascade Hall produces enriched UF_6 at a specified assay ($^{w}/_{o}$ ^{235}U), so two different assays could be produced at one time in an SBM.

The enrichment process, housed in the SBMs, is comprised of four major elements: UF_6 Feed System, Cascade System, Product Take-off System, and Tails Take-off System. Other product related functions include the Product Blending and Liquid Sampling Systems, and Contingency Dump System. Supporting functions include sample analysis, equipment decontamination and rebuild, liquid effluent treatment and solid waste management.

The major equipment used in the UF_6 feed process are Solid Feed Stations. Feed cylinders are loaded into Solid Feed Stations; vented for removal of light gases, primarily air and hydrogen fluoride (HF). The light gases and UF_6 gas generated during venting are routed to the Feed Purification Subsystem where the UF_6 is desublimed. Upon completion of venting, the feed cylinder is heated to sublime the UF_6 for use as feed gas for the centrifuges.

The major pieces of equipment in the Feed Purification Subsystem are UF_6 Cold Traps, a Vacuum Pump/Chemical Trap Sets, and a Low Temperature Take-off Stations (LTTS). The Feed Purification Subsystem removes any light gases such as air and HF from the UF_6 prior to introduction into the cascades. UF_6 is captured in UF_6 Cold Traps and ultimately recycled as feed, while HF is captured on chemical traps.

After purification, UF_6 from the Solid Feed Stations is routed to the Cascade System. Pressure in all process lines is subatmospheric.

Gaseous UF_6 from the Solid Feed Stations is routed to the centrifuge cascades. Each centrifuge has a thin-walled, vertical, cylindrically shaped rotor that spins around a central post within an outer casing. Feed, product, and tails streams enter and leave the centrifuge through the central post. Control valves, restrictor orifices, and controllers provide uniform flow of product and tails.

Depleted UF_6 exiting the cascades are transported from the high vacuum of the centrifuge for desublimation into Uranium Byproduct Cylinders (UBCs) at subatmospheric pressure. The primary equipment of the Tails Take-off System is the vacuum pumps and the Tails Low Temperature Take-off Stations (LTTS). Chilled air flows over cylinders in the Tails LTTS to effect the desublimation. Filling of the cylinders is monitored with a load cell system, and filled cylinders are transferred to an outdoor storage area (UBC Storage Pad).

Enriched UF_6 from the cascades is desublimed in a Product Take-off System comprised of vacuum pumps, Product Low Temperature Take-off Stations (LTTS), UF_6 Cold Traps, and Vacuum Pump/Chemical Trap Sets. The pumps transport the UF_6 from the cascades to the Product LTTS at subatmospheric pressure. The heat of desublimation of the UF_6 is removed by cooling air routed through the LTTS. The product stream normally contains small amounts of light gases that may have passed through the centrifuges. Therefore, a UF_6 Cold Trap and Vacuum Pump/Trap Set are provided to vent these gases from the product cylinder. Any UF_6 captured in the cold trap is periodically transferred to another product cylinder for use as product or blending stock. Filling of the product cylinders is monitored with a load cell system, and filled cylinders are transferred to the Product Liquid Sampling System for sampling.

Sampling is performed to verify product assay level ($^{w}/_{o}$ ^{235}U). The Product Liquid Sampling Autoclave is an electrically heated, closed pressure vessel used to liquefy the UF_6 and allow collection of a sample. The autoclave is fitted with a hydraulic tilting mechanism that elevates one end of the autoclave so that liquid UF_6 pours into a sampling manifold connected to the cylinder valve. After sampling, the autoclave is brought back to the horizontal position and the cylinder is indirectly cooled by water flowing through coils located on the outer shell of the autoclave.

LES customers may require product at enrichment levels other than that produced by a single Cascade Hall. Therefore, the plant has the capability to blend enriched UF_6 from two donor cylinders of different assays into a product receiver cylinder. The Product Blending System is comprised of two Blending Donor Stations and two Blending Receiver Stations, where each station can hold one 30B cylinder. The Donor Stations are similar to the Solid Feed Stations described earlier. The Receiver Station is similar to the Low-Temperature Take-off Stations described earlier.

Support functions, including sample analysis, equipment decontamination and rebuild, liquid effluent treatment and solid waste management are conducted in the Cylinder Receipt and Dispatch Building (CRDB). Decontamination, primarily of pumps and valves, uses solutions of citric acid. Sampling includes a Chemical Laboratory for verifying product UF_6 assay, and an Environmental Monitoring Laboratory (in the TSB). Liquid effluent is collected and treated and monitored before discharge to the Treated Effluent Evaporation Basin, a double-lined evaporative basin with leak detection.

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

The major structures and areas of the facility are outlined below.

<u>Separations Building Modules (SBMs)</u>

(See 12.2.1.1) The overall layout of Separations Building Module 1001 (SBM-1001) is presented in Figures 1.1-5 through 1.1-8. The overall layout of SBM-1003 is presented in Figures 1.1-9 through 1.1-12. Each SBM consists of two Cascade Halls, each having multiple cascades with each cascade having many centrifuges. The major functional areas of the SBMs are:

- Cascade Halls (2)
- Process Services Corridor
- UF_6 Handling Area

Source material and special nuclear material (SNM) are used or produced in the SBMs.

<u>Technical Services Building (TSB)</u>

(See 12.2.1.2) The overall layout of the Technical Services Building (TSB) is presented in Figures 1.1-13, Technical Services Building First Floor, and 1.1-14, Technical Services Building

Second Floor. The TSB contains support areas for the facility. It also acts as the secure point of entry to the SBMs 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.

<u>Centrifuge Assembly Building (CAB)</u>

This building is used to assemble centrifuges before they are moved into the SBMs and installed in the cascades. The overall layout of the Centrifuge Assembly Building (CAB) is presented in Figures 1.1-15 and 1.1-16. 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 (CTF)
- Centrifuge Post Mortem Facility (PMF)

Source material and SNM are used and produced in the CTF and PMF.

<u>Administration Building</u>

(See 12.2.1.6) The general office areas are located in the Administration Building. Personnel enter the Administration Building and general office areas via the main lobby.

<u>Security Building</u>

(See 12.2.1.7) 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.

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 12.2.1.3) The overall layout of the Cylinder Receipt and Dispatch Building (CRDB) is presented in Figure 1.1-17, Cylinder Receipt and Dispatch Building First Floor. The CRDB is located between two SBMs, north of the Technical Services Building. This building contains equipment to receive, inspect, weigh and temporarily store cylinders of feed UF_6 sent to the plant; temporarily store, inspect, weigh, and ship cylinders of enriched UF_6 to facility customers; receive, inspect, weigh, and temporarily store clean empty product and UBCs prior to being filled in the SBMs; 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.

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
- Staging (temporary storage) of tails and empty feed cylinders

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

- Equipment decontamination
- Rebuilding of vacuum pumps
- UF_6 cylinder valve repair
- 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 material and SNM are used in the CRDB.

Uranium Byproduct Cylinder (UBC) Storage Pad

(See 12.2.1.4) 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 UBC Storage Pad also provides buffered storage for full and empty feed cylinders. The cylinder contents are stored under vacuum in corrosion-resistant ANSI N14.1 Model 48Y cylinders. Additionally, the UBC Storage Pad provides buffered storage for clean, empty Model 30B product cylinders.

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 single girder mobile gantry crane removes the cylinders from the transporter/mover and places them in the UBC Storage Pad. The mobile gantry crane is designed to double stack the cylinders in the storage area.

Source material is used in this area.

Central Utilities Building

(See 12.2.1.5) The Central Utilities Building (CUB) is shown on Figure 1.1-18, 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.

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 molecular weight of the uranic isotopes. No chemical changes or nuclear reactions take place. The feed, product, and tails streams are all in the form of UF_6.

1.1.3.2 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 (UF_6) is the only chemical of concern that will be used at the facility. Chapter 6, Chemical Process Safety, also identifies the locations where UF_6 is stored or used in the facility and includes a detailed discussion and description of the hazardous characteristics of UF_6 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:

UF_6 Feed System

(See 12.2.2.1) The first step in the process is the receipt of the feed cylinders and preparation to feed the UF_6 through the enrichment process.

Natural UF_6 feed is received at the NEF in 48Ycylinders from a conversion plant. Pressure in the feed cylinders is below atmospheric (vacuum) and the UF_6 is in solid form.

The function of the UF_6 Feed System is to provide a continuous supply of gaseous UF_6 from the feed cylinders to the cascades. There are five[1] Solid Feed Stations per Cascade Hall.

Cascade System

(See 12.2.2.2) The function of the Cascade System is to receive gaseous UF_6 from the UF_6 Feed System and enrich the ^{235}U isotope in the UF_6 to a maximum of 5 $^{w}/_{o}$.

Multiple gas centrifuges make up arrays called cascades. The cascades separate gaseous UF_6 feed with a natural uranium isotopic concentration (0.711 $^{w}/_{o}$ ^{235}U) into two process flow streams – product and tails. The product stream is the enriched UF_6 stream, from 2 - 5 $^{w}/_{o}$$^{235}U$, with an average of 4.5 $^{w}/_{o}$$^{235}U$. The tails stream is UF_6 that has been depleted of ^{235}U isotope to 0.20 - 0.34 $^{w}/_{o}$$^{235}U$, with an average of 0.32 $^{w}/_{o}$$^{235}U$.

Product Take-off System

(See 12.2.2.3) The function of the Product Take-off System is to provide continuous withdrawal of the enriched gaseous UF_6 product from the cascades and to purge and dispose of light gas impurities from the enrichment process.

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[1] Four of the Solid Feed Stations support the current plant SWU capacity and the fifth station supports the planned SBM expansion and operational flexibility

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 UF_6 Handling Area. There are five Product LTTS per Cascade Hall.

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 UF_6 Cold Traps which capture UF_6 while leaving the light gas in a gaseous state. The cold trap is followed by product vent Vacuum Pump/Trap Sets, each consisting of a carbon trap, an alumina trap, and a vacuum pump. The carbon trap removes small traces of UF_6 and the alumina trap removes any HF from the product gas.

Tails Take-off System

(See 12.2.2.4) The primary function of the Tails Take-off System is to provide continuous withdrawal of the gaseous UF_6 tails from the cascades. A secondary function of this system is to provide a means for removal of UF_6 from the centrifuge cascades under abnormal conditions.

The tails stream exits each Cascade Hall via a primary header, goes through a pumping train, and then to Tails LTTS in the UF_6 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 12.2.2.5) The primary function of the Product Blending System is to provide a means to fill 30B cylinders with UF_6 at a specific enrichment of ^{235}U to meet customer requirements. This is accomplished by blending (mixing) UF_6 at two different enrichment levels to one specific enrichment level. The system can also be used to transfer product from a 30B cylinder to another 30B cylinder without blending.

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 12.2.2.6) 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 UF_6 in the filled product cylinders before the cylinders are sent to the fuel processor.

The Product Liquid Sampling System is one of two systems at NEF that changes solid UF_6 to liquid UF_6. The Sub-Sampling System also changes solid UF_6 to liquid UF_6.

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

The facility handles Special Nuclear Material of ^{235}U contained in uranium enriched above natural but less than or equal to the LES license limit in ^{235}U isotope. The ^{235}U is in the form of uranium hexafluoride (UF_6). 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

(See 12.2.3.3) 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

- Table 1.1-4, Estimated Annual Non-Radiological Wastes

Radioactive concentration limits and handling for liquid wastes and effluents are detailed in the Environmental Report.

The waste and effluent estimates described in the tables listed above were developed specifically for the NEF. Each system was analyzed to determine the wastes and effluents generated during operation. These values were analyzed and a waste disposal path was developed for each. LES considered the facility site, facility operation, applicable Urenco experience, applicable regulations, and the existing U.S. waste processing/disposal infrastructure during the development of the paths. The Liquid Effluent Collection and Treatment System and the Solid Waste Collection System were designed to meet these criteria.

Construction Wastes

During construction, efforts are made to minimize the environmental impact. Erosion, sedimentation, dust, smoke, noise, unsightly landscape, and waste disposal are controlled to practical levels and applicable regulatory limits. Wastes generated during site preparation and construction will be varied, depending on the activities in progress. The bulk of the wastes will consist of non-hazardous materials such as packing materials, paper and scrap lumber. These wastes will be transported off site to an approved landfill. It is estimated that the NEF will generate a non-compacted average waste volume of 3,058 m^3 (4,000 yd^3) annually.

Hazardous type wastes that may be generated during construction have been identified and annual quantities estimated are shown in Table 1.1-5, Annual Hazardous Construction Wastes. Any of these wastes that are generated will be handled by approved methods and shipped off site to approved disposal sites.

Management and disposal of all wastes from the NEF site will be performed by personnel trained to properly identify, store, and ship wastes, audit vendors, direct and conduct spill cleanup, provide interface with state agencies, maintain inventories and provide annual reports.

NEF is exempt from requiring a Spill Prevention, Control and Countermeasure Plan (SPCC). However, BMPs will be implemented during construction to minimize the possibility of spills of hazardous substances, minimize environmental impact of any spills and ensure prompt and appropriate remediation. In the event of a release, site procedures will identify individuals and their responsibilities for prompt notifications of state and local authorities. The site procedures will also identify the individuals who may determine and initiate corrective actions, if warranted.

1.2 Institutional Information

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

1.2.1 Corporate Identity

1.2.1.1 Licensee

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

Louisiana Energy Services, L.L.C.
P.O. Box 1789
275 Highway 176
Eunice, NM 88231

1.2.1.2 Organization and Management of Applicant

Louisiana Energy Services (LES), L.L.C. is a Delaware limited liability company. It has been formed solely to provide uranium enrichment services for commercial nuclear power plants. LES has one, 100% owned subsidiary, operating as a limited liability company, formed for the purpose of purchasing Industrial Revenue Bonds and no divisions. The ownership of LES is as follows:

1. Urenco Investments, Inc. (UII) (a Delaware corporation and wholly-owned subsidiary of Urenco Limited, a corporation formed under the laws of the of England ("Urenco") and owned in equal shares by Enrichment Investments Limited ("EIL"), Uranit UK Limited ("Uranit"), both companies formed under English law, and Ultra-Centrifuge Nederland NV ("UCN"), a company formed under Dutch law. EIL is ultimately wholly-owned by the government of the United Kingdom; UCN is ultimately wholly-owned by the government of the Netherlands; Uranit is ultimately owned by Eon Kenkraft GmbH (50%) and RWE Power Ag (50%), companies formed under the laws of the Federal Republic of Germany). UII holds 96% (as of December 31, 2010) of the membership units and has 100% of the voting power.

2. Urenco Deelnemingen B.V. (a Netherlands corporation and wholly-owned subsidiary of Urenco USA Inc. The ownership of Urenco USA Inc. is explicitly described above. Urenco Deelnemingen B.V. holds 4% of the membership units (as of December 31, 2010) and has 0% of the voting power.

The President of LES is Gregory OD Smith. The President reports to the Board of Managers. The Board of Managers are:

- Mr. Gregory OD Smith
  President, Chief Executive Officer
  Louisiana Energy Services, LLC
  275 Hwy 176
  Eunice , NM 88231

Mr. Smith is a citizen of the United States of America

- Dr. Helmut Engelbrecht
  Chief Executive Officer
  Urenco Limited
  18 Oxford Road
  Marlow Bucks
  SL7 2NL, United Kingdom

  Dr. Engelbrecht is a citizen of the Federal Republic of Germany

- Mr. Friso van Oranje
  Chief Financial Officer
  Urenco Limited
  18 Oxford Road
  Marlow Bucks
  SL7 2NL, United Kingdom

  Mr. van Oranje is a citizen of the Netherlands

- Dr. Charles W. Pryor, Jr.
  Chairman of the Board
  Urenco USA, Inc.
  1164 Pryor Ridge Trail Lynchburg, VA24503
  Dr. Pryor is a citizen of the United States of America

- Mr. Christopher Chater
  Board of Managers Member
  Bahnhofstrasse 8
  48455 Bad Bentheim
  Germany

  Mr. Chater is a citizen of the United Kingdom

The Vice President - Operations is the primary regulatory contact and is responsible for the safe operation of the URENCO USA Facility. LES' principal location for business is Eunice, New Mexico. The facility will be located in Lea County near Eunice, New Mexico. No other companies will be present or operating on the URENCO USA site other than services specifically contracted by LES.

Foreign Ownership, Control and Influence (FOCI) of LES is addressed in the URENCO USA Standard Practice Procedures for the Protection of Classified Matter, Appendix 1 – FOCI Package. The NRC in their letter dated, March 24, 2003, has stated "...that while the mere presence of foreign ownership would not preclude grant of the application, any foreign relationship must be examined to determine whether it is inimical to the common defense and security [of the United States]". (NRC, 2003) The FOCI Package mentioned above, and amendments thereto, provide sufficient information for this examination to be conducted.

1.2.1.3 Address of the Enrichment Plant and Legal Site Description

The URENCO USA site is physically located approximately 8 km (5 mi) east of Eunice, New Mexico adjacent to New Mexico Highway 234 in Lea County. The legal description is as follows:

A PARCEL OF LAND WITHIN SECTION 32, TOWNSHIP 21 SOUTH, RANGE 38 EAST, NEW MEXICO PRINCIPAL MERIDIAN, LEA COUNTY, NEW MEXICO,

BEGINNING at the one-quarter corner between Sections 31 and 32, (a found GLO brass cap on a 2-in iron pipe);

THENCE N00°38'22"W along the section line between Sections 31 and 32 a distance of 2638.37 feet to the corner of Sections 29, 32, 31 and 30, (a found GLO brass cap on a 2-in iron pipe);

THENCE N89°18'08"E along the section line between Sections 29 and 32 a distance of 2640.69 feet to a set 5/8-in rebar with a 2-in aluminum cap marked "MUTH PLS 13239";

THENCE N89°18'08"E along the section line between Sections 29 and 32 a distance of 2640.69 feet to the corner of Sections 28, 33, 32 and 29, (a found GLO brass cap on a 2-in iron pipe);

THENCE S00°39'20"E along the section line between Sections 32 and 33 a distance of 2640.49 feet to the one-quarter corner between Sections 32 and 33, (a found GLO brass cap on a 1-in iron pipe);

THENCE S00°41'56"E along the section line between Sections 32 and 33 a distance of 2324.52 feet to a found railroad iron marking the right-of-way for New Mexico State Highway No. 234; from whence the corner of Sections 33 and 32 of Township 21 South, Range 38 East, and Sections 4 and 5 of Township 22 South, Range 38 East (a found 1/2-in rebar) bears S00°41'56"E a distance of 340.08 ft;

THENCE N80°10'49"W along the observed northerly right-of-way line of New Mexico State Highway No. 234 a distance of 5377.12 ft to a point of intersection with the section line between Sections 31 and 32 (set 5/8-in rebar with a 2-in aluminum cap marked "MUTH PLS 13239"); from whence the corner of Sections 31 and 32 of Township 21 South, Range 38 East, and Sections 6 and 5 of Township 22 South, Range 38 East (a found GLO brass cap on a 2-in iron pipe) bears S00°35'16"E a distance of 1321.66 ft;

THENCE N00°35'16"W along the section line between Sections 31 and 32 a distance of 1345.14 to the POINT OF BEGINNING

Said Parcel CONTAINS 542.80 ACRES more or less

1.2.2 Financial Information

LES estimates the total cost of the URENCO USA site to be approximately $1.2 billion (in 2002 dollars), excluding escalation, contingency, interest, tails disposition, decommissioning, and any replacement equipment required during the life of the facility.

1.2 Institutional Information

There are financial qualifications to be met before a license can be issued. LES acknowledges the use of the following Commission-approved criteria as described in Policy Issues Associated with the Licensing of a Uranium Facility; Issue 3, Financial Qualifications (LES, 2002) in determining if the project is financially feasible:

1. Construction of the facility shall not commence before funding (except decommissioning funding, and liability insurance as discussed below) is fully committed. Of this full funding (equity and debt), the applicant must have in place before constructing the associated capacity: (a) a minimum of equity contributions of 30% of project costs from the parents; and (b) firm commitments ensuring funds for the remaining project costs.
2. LES shall not proceed with the project unless it has in place long-term enrichment contracts (i.e., five years) with prices sufficient to cover both construction and operation costs, including a return on investment, for the entire term of the contracts.
3. In accordance with the approved Exemption from certain provisions of 10 CFR 40.36 as discussed in Section 1.2.5 of this SAR, decommissioning funding will be provided incrementally. Therefore, receipt of UF_6 into a building shall not commence before the final executed copies of the reviewed financial assurance instruments for that building are provided to the NRC.

LES shall in accordance with 10 CFR 140.13b, (CFR, 2003l), prior to and throughout operation, have and maintain nuclear liability insurance in the type and amounts the Commission considers appropriate up to a limit of $300 million to cover liability claims arising out of any occurrence within the United States, causing, within or outside the United States, bodily injury, sickness, disease, or death, or loss of or damage to property, or loss of use of property, arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source or special nuclear material.

The amounts of nuclear energy liability insurance required may be furnished and maintained in the form of:

1. An effective facility form (non-indemnified facility) policy of nuclear energy liability insurance from American Nuclear Insurers and/or Mutual Atomic Energy Liability underwriters; or
2. Such other type of nuclear energy liability insurance as the Commission may approve; or
3. A combination of the foregoing.
4. $5 million to receive and maintain onsite, an inventory of ≤ 50 kg of natural or depleted UF_6 as "test material".
5. $300 million to receive and maintain onsite, an inventory > 50 kg of UF_6 on site as "feed material".

If the form of liability insurance will be other than an effective facility form (non-indemnified facility) policy of nuclear energy liability insurance from American Nuclear Insurers and/or Mutual Atomic Energy Liability Underwriters, such form will be provided to the Nuclear Regulatory Commission by LES. The effective date of this incremental insurance will be no later than the date that LES takes possession of the above specified quantity and enrichment of UF_6.

Effective November 26, 2002, nuclear energy liability Facility Form policy number NF-0350 was issued to LES for the planned NEF with the limit of liability of $1,000,000. This standby limit will apply until the plant takes possession of UF_6 in a quantity listed in #4 or #5 above, at which time it is anticipated that the liability insurance coverage limit will be increased to $5 million for "test material", or the $300 million limit for quantities of UF_6 in excess of the 50 kg "test material" limit. Until such time as LES takes possession of source material UF_6, the effects described in 10 CFR 140.13b involving source material are not possible. Therefore, the $1,000,000 standby liability policy, in addition to appropriate construction coverage, is considered to be sufficient for the construction phase. LES will provide proof of liability insurance of a type and in the amounts to cover liability claims required by 10 CFR 140.13b prior to taking possession of source material.

Information indicating how reasonable assurance will be provided that funds will be available to decommission the facility as required by 10 CFR 70.22(a)(9) (CFR, 2003b), 10 CFR 70.25 (CFR, 2003c), and 10 CFR 40.36 (CFR, 2003d) is described in detail in Chapter 10, Decommissioning.

1.2.3 Type, Quantity, and Form of Licensed Material

LES is licensed to acquire, deliver, receive, possess, produce, use, transfer, and/or store special nuclear material (SNM) meeting the criteria of special nuclear material of low strategic significance as described in 10 CFR 70.4 (CFR, 2003e). Details are provided in Table 1.2-1, Type, Quantity, and Form of Licensed Material. Byproduct materials and selected SNM sources are presented in the current version of SNM-2010.

1.2.4 Requested Licenses and Authorized Uses

LES is engaged in the production and selling of uranium enrichment services to electric utilities for the purpose of manufacturing fuel to be used to produce electricity in commercial nuclear power plants.

This application is for the necessary licenses issued under 10 CFR 70 (CFR, 2003f), 10 CFR 30 (CFR, 2003g) and 10 CFR 40 (CFR, 2003h) to construct, own, use and operate the facilities described herein as an integral part of the uranium enrichment facility. This includes licenses for source, special nuclear material and byproduct material. The period of time for which the license is requested is 30 years.

See Section 1.1, Facility and Process Description for a summary, non-technical narrative description of the enrichment activities utilized in NEF.

1.2.5 Special Exemptions or Special Authorizations

In accordance with 10 CFR 40.14 (CFR, 2005a), "Specific exemptions," and 10 CFR 70.17 (CFR, 2005b), "Specific exemptions," LES requests exemptions from certain provisions of 10 CFR 40.36 (CFR, 2005c), "Financial assurance and recordkeeping for decommissioning," paragraph (d), and 10 CFR 70.25 (CFR, 2005d), "Financial assurance and recordkeeping for decommissioning," paragraph (e). Specifically, 10 CFR 40.36(d) (CFR, 2005c) and

10 CFR 70.25(e) (CFR, 2005d) both state in part that "...the decommissioning funding plan must also contain a certification by the licensee that financial assurance for decommissioning

has been provided in the amount of the cost estimate for decommissioning...." As stated in Section 10.2.1, "Decommissioning Funding Mechanism," of the SAR since LES intends to sequentially install and operate modules of the enrichment equipment over time, providing financial assurance for decommissioning during the operating life of the NEF at a rate that is in proportion to the decommissioning liability for these facilities as they are phased in satisfies the requirements of this regulation without imposing the financial burden of maintaining the entire financial coverage for facilities and material that are not yet in existence. The same basis applies to decommissioning funding assurance for depleted uranium byproduct. As also stated in Section 10.2.1 of the SAR, LES proposes to provide financial assurance for the disposition of depleted uranium byproduct at a rate in proportion to the amount of accumulated depleted uranium byproduct onsite up to the maximum amount of the depleted uranium byproduct produced by the NEF.

The justification for this proposal to provide decommissioning funding assurance on a forward-looking incremental basis is LES' commitment to update the decommissioning cost estimates and to provide to the NRC a revised funding instrument for facility decommissioning at a minimum prior to the operation of each facility module. With respect to the depleted uranium byproduct, LES commits to updating the decommissioning cost estimates on an annual forward-looking incremental basis and to providing the NRC revised funding instruments that reflect these projections of depleted uranium byproduct production. The long-term nature of enrichment contracts allows LES to accurately predict the production of depleted uranium byproduct. If any adjustments to the funding assurance were determined to be needed during the annual period due to production variations, they would be made promptly and a revised funding instrument would be provided to the NRC.

LES requests that exemptions from the provisions of 10 CFR 40.36(d) (CFR, 2005c) and 10 CFR 70.25(e) (CFR, 2005d) described above be granted. In support of this request, LES provides the following information relative to the criteria in 10 CFR 40.14 (CFR, 2005a) and 10 CFR 70.17 (CFR, 2005b).

Granting the exemption is authorized by law

There is no statutory prohibition to providing decommissioning funding assurance on an incremental basis. In fact, the NRC has previously accepted an incremental approach to decommissioning funding assurance for the United States Enrichment Corporation's operation of its gaseous diffusion plants.

Granting the exemptions will not endanger life or property or the common defense and security

Allowing the decommissioning funding assurance for the NEF to be provided on a forward-looking incremental basis continues to ensure that adequate funds are available at any point in time after licensed material is introduced onto the NEF site to decommission the facility and disposition any depleted uranium byproduct possessed by LES. Accordingly, life, property, or the common defense and security will not be endangered by the NEF once it is permanently shutdown.

Granting the exemptions is otherwise in the public interest

Providing an alternative, diverse, and secure domestic source of enrichment services in support of the nuclear power industry that supplies 20% of the nation's electricity is clearly in the public benefit. Providing decommissioning funding assurance on an incremental basis will ensure that adequate financial assurance is available when required. Imposing the requirement to provide decommissioning funding assurance for the entire facility and all depleted uranium byproduct that would be produced over the NEF licensed operating period results in a significant unnecessary financial hardship. Accordingly, the granting of these exemptions is in the public interest.

Since the granting of this exemption does not satisfy any of the criteria for categorical exclusion delineated in 10 CFR 51.22 (CFR, 2005e), "Criteria for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," nor the criteria requiring an environmental impact statement in

10 CFR 51.20 (CFR, 2005f), "Criteria for and identification of licensing and regulatory actions requiring environmental impact statements," an environmental assessment is required in accordance with 10 CFR 51.21 (CFR, 2005g), "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." Accordingly, LES proposes that the NRC make a finding of no significant impact based on the following information addressing the provisions of 10 CFR 50.30 (CFR, 2005h), "Environmental assessment."

<u>Need for the proposed action</u>

Granting of the requested exemption will allow LES to satisfy the applicable decommissioning funding assurance requirements for the NEF without imposing an unnecessary financial burden on LES.

<u>Alternatives as required by Section 102(2)(E) of the National Environmental Policy Act (NEPA)</u>

The only alternative to granting the requested exemption is to not grant it. The significant financial burden that would be imposed on LES by not granting the requested exemption is unnecessary.

<u>The environmental impacts of the proposed action and alternatives as appropriate</u>

Granting the requested exemption will not result in environmental impacts in addition to those delineated in the ER for the NEF since adequate funds will continue to be available to decommission the NEF and disposition any depleted uranium byproduct possessed by LES at any point in time after licensed material is introduced onto the NEF site. The environmental impact of not granting the requested exemption could potentially be the loss of an alternate, diverse, and secure domestic source of enrichment services for the nuclear power industry that supplies 20% of the nation's electricity.

<u>A list of agencies and persons consulted and identification of sources used</u>

The NRC Project Manager for the NEF was contacted. The NEF license application was used as a source.

Based on the above information, LES proposes that, if this exemption request is granted, the NRC reach a finding of no significant impact in accordance with 10 CFR 51.32 (CFR, 2005i), "Finding of no significant impact."

1.2.6 Security of Classified Information

Access to restricted data or national security information shall be controlled in accordance with 10 CFR 10 (CFR, 2003i), 25 (CFR, 2003j), and 95 (CFR, 2003k). This application does contain classified information that has been submitted under separate correspondence.

1.3 Site Description

The NEF is located in southeastern New Mexico in Lea County near the border of Andrews County, Texas. The site consists of land north of New Mexico Highway 234 within Section 32 of Township 21 S, Range 38 E. The nearest communities are Eunice, about 8 km (5 mi) due west and Hobbs about 32 km (20 mi) north of the site. The area surrounding the site consists of vacant land and industrial properties. A railroad spur borders the site to the north. Further north is a sand/aggregate quarry operated by the Wallach Concrete Company. The quarry owner leases land space to a "produced water" reclamation company, Sundance Services, which maintains three small "produced water" lagoons. There is also a man-made pond stocked with fish on the quarry property.

A vacant parcel of land, Section 33, is immediately to the east. Section 33 borders the New Mexico/Texas state line that is 0.8 km (0.5 mi) east of the site. Several disconnected power poles are situated in front of Section 33, parallel to New Mexico Highway 234. Land further east, in Texas, is occupied by Waste Control Specialists (WCS), LLC. WCS possesses a radioactive materials license from Texas, an NRC Agreement state, and is licensed to treat and temporarily store low-level radioactive waste. Land east of WCS is occupied by the Letter B Ranch.

High power utility lines run in a north-south direction near the property line of WCS, parallel to the New Mexico/Texas state line.

To the southeast, across New Mexico Highway 234, is the Lea County Landfill.

Land further north, south and west has mostly been developed by the oil and gas industry.

An underground CO2 pipeline owned by Trinity Pipeline, LLC, originally running southeast-northwest, now relocated to north south at the western boundary traverses the property. An underground natural gas pipeline owned by the Sid Richardson Energy Services Company is located along the south property line, paralleling New Mexico Highway 234.

An active railroad line, operated by the Texas-New Mexico Railroad, runs parallel to New Mexico Highway 18 and just east of Eunice within 8 km (5 mi) of Section 32. There is also an active railroad spur that runs from the Texas-New Mexico Railroad line, along the north boundary of Section 32 and terminates at the WCS facility.

Figure 1.3-1, Five Mile Radius, Radial Sectors, shows the physical features surrounding the facility to an 8 km (5 mi) radius.

1.3.1 Site Geography

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

1.3.1.1 Site Location Specifics

The proposed 220 ha (543 acre) site is located within Section 32 of Township 21 S in southeastern New Mexico in Lea County approximately 0.8 km (0.5 mi) west of the Texas state border, 51 km (32 mi) west-north-west of Andrews, Texas and 523 km (325 mi) southeast of Albuquerque, New Mexico. This location is 8 km (5 mi) due east of Eunice and 32 km (20 mi) south of Hobbs. The geographic location of the facility is shown on Figures 1.1-1, State Map, and 1.1-2, County Map.

The approximate center of the NEF is at latitude 32 degrees, 26 minutes, 1.74 seconds North and longitude 103 degrees, 4 minutes, 43.47 seconds West. Section 32 is currently owned by the State of New Mexico and is being acquired by LES through a state land swap arrangement. Until the land swap is completed, LES has been granted a 35 year easement by the State of New Mexico for site access and control.

Figure 1.1-4, Facility Layout (Site Plan) with Site Boundary and Controlled Access Area Boundary, shows the site property boundary, including the Controlled Access Area and the general layout of the buildings.

1.3.1.2 Features of Potential Impact to Accident Analysis

The NEF site is located in the Pecos Plains Section of the Great Plains Province. Site topography is relatively level, with an overall gradual rise in elevation from the southwest to the northeast. An area comprised of small sand hills exists along the west property line. There are no mountain ranges in the immediate vicinity. Earthquakes in the region are isolated or occur in small clusters of low to moderate size events toward the Rio Grande Valley of New Mexico and southeast of the NEF site in Texas.

An underground natural gas pipeline owned by the Sid Richardson Energy Services Company is located along the south property line, paralleling New Mexico Highway 234.

An underground CO2 pipeline owned by Trinity Pipeline, LLC, running southeast-northwest, originally traversed the property. This pipeline has been relocated to the western edge of the NEF site property boundary.

New Mexico Highway 234 runs parallel to the southern property line. New Mexico Highway 234 intersects New Mexico Highway 18 about 4 km (2.5 mi) to the west.

An active railroad line operated by the Texas-New Mexico Railroad runs parallel to Highway 18 and just east of Eunice within 8 km (5 mi) of Section 32.

1.3.2 Demographics

This section provides the census results for the facility site area, and includes specific information about populations, public facilities (schools, hospitals, parks, etc.) and land and water use near the site.

1.3.2.1 Latest Census Results

The combined population of the two counties in the NEF vicinity, based on the 2000 U.S. Census is 68,515, which represents a 2.3% decrease from the 1990 population of 70,130. This decrease is counter to the trends for the states of New Mexico and Texas which had population increases of 20.1% and 22.8%, respectively during the same decade. Over that 10 year period, Lea County, New Mexico, where the site is located, had a growth decrease of 0.5%. The growth decrease in Andrews County, Texas was 9.3%. Lea County experienced a sharp but short population increase in the mid-1980's due to an influx of petroleum industry jobs. That influx caused its population to increase to over 65,000 during that period.

Based on projections made using historic data, the population of Lea County, New Mexico and Andrews County, Texas is likely to grow more slowly than their respective states over the next 30 years (the anticipated license period of the NEF).

Based on U. S. census data the minority populations of the Lea County New Mexico and Andrews County Texas as of 2000 were 32.9% and 22.9%, respectively. These percentages are consistent with their respective state averages of 34.7% and 26.4%.

The low income population of Lea County, New Mexico and Andrews County, Texas are 21.1% and 16.4% respectively. These percentages are consistent with their respective state averages of 18.4% and 15.4%. Within the site area the percentage of population below the poverty level is significantly lower in both states.

1.3.2.2 Description, Distance, And Direction To Nearby Population Areas

The NEF site is in Lea County, New Mexico near the border of Andrews County, Texas. The nearest community is Eunice, approximately 8 km (5 mi) east of the site. Other population centers are at distances from the site as follows:

- Hobbs, Lea County, New Mexico: 32 km (20 mi north)
- Jal, Lea County, New Mexico: 37 km (23 mi south)
- Lovington, Lea County New Mexico: 64 km (39 mi north-northwest)
- Andrews, Andrews County Texas: 51 km (32 mi east)
- Seminole, Gaines County Texas: 51 km (32 mi east-northeast)
- Denver City, Gaines County, Texas: 65 km (40 mi) north-northeast

Aside from these communities, the population density around the site is extremely low. The nearest large population center (>100,000) is Midland-Odessa, Texas which is approximately 103 km (64 mi) to the southeast.

1.3.2.3 Proximity to Public Facilities – Schools, Hospitals, Parks

The Eunice First Assembly of God Church is located about 9 km (5.4 mi) from the site.

There are two hospitals in the vicinity of the site. The Lea Regional Medical Center is located in Hobbs, New Mexico about 32 km (20 mi) north of the NEF site. This 250-bed hospital can handle acute and stable chronic care patients. In Lovington, New Mexico, 64 km (39 mi) north-northwest of the site, Covenant Medical Systems manages Nor-Lea Hospital, a full-service, 27-bed facility.

Eunice Senior Center is located about 9 km (5.4 mi) from the site.

There are four educational facilities within about 8 km (5 mi) of the NEF site, all in Eunice, New Mexico. These include an elementary school, a middle school, a high school, and a private K-12 school.

Eunice Fire and Rescue and the Eunice Police Department are located approximately 8 km (5 mi) from the site.

The Eunice Golf Course is located approximately 14.7 km (9.4 mi) from the site.

1.3.2.4 Nearby Industrial Facilities (Includes Nuclear Facilities)

<u>Nuclear Facilities</u>

There are no nuclear production facilities located within 32 km (20 mi) of the site, therefore neither environmental nor emergency preparedness interactions between facilities is required.

<u>Non-Nuclear Facilities</u>

The site is bordered to the north by railroad tracks beyond which is a quarry operated by Wallach Concrete Company. The quarry owner leases land space to Sundance Services, a reclamation company that maintains three small “produced water” lagoons.

Lea County operates a landfill on the south side of Section 33 across New Mexico State Highway 234, approximately 1 km (0.6 mi) from the center of the site.

A vacant parcel of land is immediately east of the site. Land further east, in Texas, is occupied by WCS. WCS possesses a radioactive materials license from Texas, an NRC Agreement state, and is licensed to treat and temporarily store low-level radioactive waste.

Dynegy’s Midstream Services Plant is located 6 km (4 mi) from the site. This facility is engaged in the gathering and processing of natural gas for the subsequent fractionation, storage, and transportation of natural gas liquids.

An underground CO2 pipeline, running southeast-northwest, originally traversed the property. This underground CO2 pipeline has been relocated to the western edge of the property boundary.

An underground natural gas pipeline is located along the south property line, paralleling New Mexico Highway 234.

Eunice maintains water supply tanks approximately 8 km (5 mi) north and 8 km (5 mi) south of the site.

Land further north, south and west of the site has mostly been developed by the oil and gas industry.

The Eunice Airport is situated about 8 km (5 mi) west of the town center. The nearest commercial carrier airport is Lea County Regional Airport in Hobbs, New Mexico about 40 km (25 mi) north-northwest of the site. A major commercial airport in Midland-Odessa, Texas is approximately 103 km (64 mi) to the southeast.

1.3.2.5 Land Use Within Eight Kilometers (Five Mile) Radius, Uses Of Nearby Bodies Of Water

The site and vicinity are within the southern part of the Llano Estacado or Staked Plains, which is a remnant of the Southern High Plains. The site area overlies prolific oil and gas geologic formations of the Pennsylvanian and Permian age.

Onsite soils consist of fine sand, loamy fine sand and loose sands surrounding large barren sand dunes and are common to areas used for rangeland and wildlife habitat.

Surrounding property consists of vacant land and industrial developments. Gas and oil field operations are widespread in the area, but significant petroleum potential is absent within 5 to 8 km (3 to 5 mi) of the site.

More than 98% of the area within an 8 km (5 mi) radius of the NEF is an extensive area of open land on which livestock wander and graze. Built-up land (1.2%) and barren land (0.3%) constitute the other two land use classifications in the site vicinity.

Baker Spring, an intermittent surface water feature, is situated a little over 1.6 km (1 mi) northeast of the NEF site.

The facility will make no use of either surface water or groundwater supply from the site. A site Stormwater Detention Basin will discharge to the ground and a site sewer system will send sanitary wastewater to the City of Eunice Wastewater Treatment Plant with a Groundwater Discharge Permit/Plan from the New Mexico Water Quality Bureau. Six septic tanks, each with one or more leach fields, may be installed as a backup to the sanitary waste system. No significant adverse changes are expected in site hydrology as a result of construction or operation of the NEF. Section 4, Environmental Impacts, of the Environmental Report addresses potential for impacts on site hydrology as a result of activities on the site.

1.3.3 Meteorology

In this section, data characterizing the meteorology (e.g., winds, precipitation, and severe weather) for the site are presented.

1.3.3.1 Primary Wind Directions And Average Wind Speeds

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

Meteorological data from the National Weather Service (NWS) site at Midland-Odessa, Texas, indicate an annual mean wind speed of 4.9 m/s (11.0 mi/hr). The prevailing wind direction is wind from the south. The maximum five-second wind speed is 31.3 m/sec (70 mph) from 200 degrees with respect to true north.

By comparison, the data from Roswell, New Mexico indicate the annual mean wind speed is 3.7 m/s (8.2 mi/hr) and the prevailing wind direction is wind from the south-southeast. The maximum five-second wind speed is 27.7 m/sec (62 mph) from 270 degrees with respect to true north.

These and additional data are discussed and further analyzed in the Environment Report.

1.3.3.2 Annual Precipitation – Amounts and Forms

The NEF site is located in the Southeast Plains of New Mexico near the Texas border. The climate is typical of a semi-arid region, with generally mild temperatures, low precipitation and humidity, and a high evaporation rate. Vegetation consists mainly of native grasses and some mesquite trees. During the winter, the weather is often dominated by a high-pressure system located in the central part of the western United States and a low-pressure system located in north-central Mexico. During the summer, the region is affected by a low-pressure system normally located over Arizona.

The normal annual total rainfall as measured in Hobbs, New Mexico is 46.1 cm (18.15 in). Precipitation amounts range from an average of 1.22 cm (0.48 in) in March to 7.95 cm (3.13 in) in September. Record maximum and minimum monthly totals are 35.13 cm (13.83 in) and zero respectively. (WRCC, 2003)

The normal annual total rainfall in Midland-Odessa, Texas, is 37.6 cm (14.8 in). Precipitation amounts range from an average of 1.1 cm (0.42 in) in March to 5.9 cm (2.31 in) in September. Record maximum and minimum monthly totals are 24.6 cm (9.70 in) and zero, respectively. The highest 24-hour precipitation total was 15.2 cm (5.99 in) in July 1968 (NOAA, 2002a).

The normal annual rainfall total as measured in Roswell, New Mexico, is 33.9 cm (13.34 in). Record maximum and minimum monthly totals are 17.50 cm (6.88 in) and zero , respectively (NOAA, 2002b, 2002a). The highest 24-hour precipitation total was 12.47 cm (4.91 in) in July 1981 (NOAA, 2002b).

Snowfall in Midland-Odessa, Texas, averages 13.0 cm (5.1 in) per year. Maximum monthly snowfall/ice pellets of 24.9 cm (9.8 in) fell in December 1998. The maximum amount of snowfall/ice pellets to fall in 24 hours was 24.9 cm (9.8 in) in December 1998 (NOAA, 2002a).

Snowfall in Roswell, New Mexico averages 30.2 cm (11.9 in) per year. Maximum monthly snowfall/ice pellets of 53.3 cm (21.0 in) fell in December 1997. The maximum amount of snowfall/ice pellets to fall in 24 hours was 41.91 cm (16.5 in) in February 1988 (NOAA, 2002b).

Additional details on rainfall and snowfall are provided in the Environmental Report.

The design basis ground snow load was developed using the methodology prescribed in the NRC Site Analysis Branch Position for Winter Precipitation Loads (NRC, 1975). The prescribed load to be included in the combination of normal live loads is based on the weight of the 100 year snowfall or snowpack whichever is greater. The winter precipitation load to be included in the combination of extreme live loads is based on the sum of the weight of the 100 year snowpack and the weight of the 48 hour Probable Maximum Winter Precipitation (PMWP) for the month corresponding to the selected snowpack.

The 100 year mean recurrence ground snow load was calculated to be 58.5 kg/m2 (12 lb/ft2), and the applicable PMWP was calculated to be 96.6 kg/m2 (19.8 psf). The addition of these two figures results in a design load of 155.1 kg/m2 (32 lb/ft2).

1.3.3.3 Severe Weather

Tornadoes

Tornadoes occur infrequently in the vicinity of the NEF. Only two tornadoes were reported in Lea County, New Mexico, (Grazulis, 1993) from 1880-1989. Across the state line, only one tornado was reported in Andrews County, Texas, (Grazulis, 1993) from 1880-1989.

Tornadoes are commonly classified by their intensities. The F-Scale classification of tornados is based on the appearance of the damage that the tornado causes. There are six classifications, F0 to F5, with an F0 tornado having winds of 61-116 km/hr (40-72 mi/hr) and an F5 tornado having winds of 420-520 km/hr (261-318 mi/hr) (AMS, 1996). The two tornadoes reported in Lea County were estimated to be F2 tornadoes (Grazulis, 1993).

The design parameters applicable to the design tornado with a period of recurrence of 100,000 years are as follows:

Design Wind Speed	302 km/hr	188 mi/hr
Radius of damaging winds	130 m	425 ft
Atmospheric pressure change (APC)	-390 kg/m^2	-80 lb/ft^2
Rate of APC	-146 $kg/m^2/s$	-30 $lb/ft^2/s$

Hurricanes

Hurricanes, or tropical cyclones, are low-pressure weather systems that develop over the tropical oceans. Hurricanes are fueled by the relatively warm tropical ocean water and lose their intensity quickly once they make landfall. Since the NEF is located about 805 km (500 mi) from the coast, it is most likely that any hurricane that tracked towards the site would have dissipated to the tropical depression stage, that is, wind speeds less than 63 km/hr (39 mi/hr), before it reached the NEF. Hurricanes are therefore not considered a threat to the NEF.

Thunderstorms and Lightning Strikes

Thunderstorms occur during every month but are most common in the spring and summer months. Thunderstorms occur an average of 36.4 days/year in Midland/Odessa (based on a 54-year period of record (NOAA, 2002a). The seasonal averages are: 11 days in spring (March through May); 17.4 days in summer (June through August); 6.7 days in fall (September through November); and 1.3 days in winter (December through February).

The current methodology for estimating lightning strike frequencies includes consideration of the attractive area of structures (Marshall, 1973). This method consists of determining the number of lightning flashes to earth per year per square kilometer and then defining an area over which the structure can be expected to attract a lightning strike.

Using this methodology, the attractive area of the facility structures has been conservatively determined to be 0.071 km^2. Using 4 flashes to earth per year per square kilometer (2.1 flashes to earth per year per square mile) (NWS, 2003b) it can be estimated that the NEF will experience approximately 1.36 flashes to earth per year.

Sandstorms

Blowing sand or dust may occur occasionally in the area due to the combination of strong winds, sparse vegetation, and the semi-arid climate. High winds associated with thunderstorms are frequently a source of localized blowing dust. Dust storms that cover an extensive region are rare, and those that reduce visibility to less than 1.61 km (1 mile) occur only with the strongest pressure gradients such as those associated with intense extratropical cyclones which occasionally form in the area during winter and early spring (DOE, 2003).

1.3.4 Hydrology

The hydrology information presented for the NEF was based on a subsurface investigation initiated at the NEF site in September 2003. Extensive subsurface investigations for a nearby facility, WCS, located to the east of the NEF site, have also provided hydrogeologic data that was used in planning the NEF surface investigation. Other literature searches were also conducted to obtain reference material.

The NEF site itself contains no surface water bodies or surface drainage features. Essentially all the precipitation that occurs at the site is subject to infiltration and/or evapotranspiration. Groundwater was encountered at depths of 65 to 68 m (214 to 222 ft). Significant quantities of groundwater are only found at depths over 340 m (1,115 ft) where cover for that aquifer is provided by 323 to 333 m (1,060 to 1,092 ft) or more of clay.

1.3.4.1 Characteristics Of Nearby Rivers, Streams, And Other Bodies Of Water

The climate in southeast New Mexico is semi-arid. Precipitation averages only 33 to 38 cm (13 to 15 in) a year. Evaporation and transpiration rates are high. This results in minimal, if any surface water occurrence or groundwater recharge.

The NEF site contains no surface drainage features, such as arroyos or buffalo wallows. The site topography is relatively flat. Some localized depressions exist, due to eolian processes, but the size of these features is too small to be of significance with respect to surface water collection.

1.3.4.2 Depth To The Groundwater Table

The site subsurface investigation performed during September 2003 had two main objectives:

1) to delineate the depth to the top of the Chinle Formation red bed clay that exists beneath the NEF site to assess the potential for saturated conditions above the red beds, and 2) to complete three monitoring wells in the siltstone layer beneath the red beds to monitor water level and water quality within this thin horizon of perched intermittent saturation. The presence of the thick Chinle clay beneath the site essentially isolates the deep and shallow hydrologic systems. Groundwater occurring within the red bed clay occurs at three distinct and distant elevations. Approximately 65 to 68 m (214 to 222 ft) beneath the land surface, within the red bed unit, is a siltstone or silty sandstone unit with some saturation. Because it is a low permeability formation that does not yield groundwater very readily it is not considered to be an aquifer. This siltstone layer is hydraulically isolated from the near surface hydrologic conditions due to the presence of a thick clay sequence above it.

The next water bearing unit below the saturated siltstone horizon is a saturated 30.5-meter (100 foot) thick sandstone horizon approximately 183 m (600 ft) below land surface, which overlies the Santa Rosa formation. The Santa Rosa formation is the third water bearing unit and is located about 340 m (1,115 ft) below land surface. Between the siltstone and sandstone saturated horizons and the Santa Rosa formation lie a number of layers of sandstones, siltstones, and shales. Hydraulic connection between the siltstone and sandstone saturated horizons and the Santa Rosa formation is non-existent.

No withdrawals or injection of groundwater will be made as a result of operation of the NEF facility. Thus, there will be no affect on any inter-aquifer water flow.

1.3.4.3 Groundwater Hydrology

The climate in southeast New Mexico is semi-arid, and evapotranspiration processes are significant enough to short-circuit any potential groundwater recharge. There is some evidence for shallow (near-surface) groundwater occurrence in areas to the north at the Wallach Concrete plant. These conditions are intermittent and limited. The typical geologic cross section at that location consists of a layer of caliche at the surface, referred to as the "caprock." In some areas the caprock is missing and the sand and gravel are exposed at the surface. The caprock is generally fractured and, following precipitation events may allow infiltration that quickly bypasses any roots from surface vegetation. In addition, there are areas where the sand and gravel outcrop may allow rapid infiltration of precipitation. These conditions have led to instances of minor amounts of perched groundwater at the base of the sand and gravel unit, atop the red beds of the Chinle Formation.

Conditions at the NEF site are different than at the Wallach Concrete site. The caprock is not present at the NEF site. Therefore, rapid infiltration through fractured caliche does not contribute to localized recharge at the NEF site.

Another instance of possible saturation above the Chinle clay may be seen at Baker Spring, just to the northeast of the NEF site where the caprock ends. The surface water is intermittent, and water typically flows from Baker Spring only after precipitation events. Some water may seep from the sand and gravel unit beneath the caprock, but deep infiltration of water is impeded by the low permeability of the Chinle clay in the area. This condition does not exist at the NEF site due to the absence of the caprock and the low permeability surface soils.

A third instance of localized shallow groundwater occurrence exists to the east of the NEF site where several windmills on the WCS property were formerly used to supply water for live stock tanks. These windmills tapped small saturated lenses above the Chinle Formation red beds, but the amount of groundwater in these zones was limited.

1.3.4.4 Characteristics Of The Uppermost Aquifer

The first occurrence of a well-defined aquifer is approximately 340 m (1,115 ft) below land surface, within the Santa Rosa formation. No impacts are expected to the aquifer from the NEF because of the depth of the Santa Rosa formation, the thick Chinle clay overburden, and the fact that the NEF will not consume surface or groundwater or discharge to the surrounding area.

Treated liquid effluents are discharged to the onsite Treated Effluent Evaporative Basin, a double-lined evaporative basin with leak detection.

1.3.4.5 Design Basis Flood Events Used For Accident Analysis

The closest water conveyance is Monument Draw; a typically dry, intermittent stream located about 4 km (2.5 mi) west of the site. Since there are no bodies of water in the immediate vicinity of the site, flood is not a design basis event for the NEF. Additionally a diversion ditch is strategically located to deflect surface runoff from adjacent land away from the facility structures on the site.

The only potential flooding of the plant results from local intense rainfall. Flood protection against the local Probable Maximum Precipitation (PMP) is provided by establishing the facility floor level above the calculated depth of ponded water caused by the local PMP. The CUB contains a sub-floor level cable spreading room. Access to the cable spreading room is via enclosed ladders at either end of that room.

1.3.5 Geology

This section provides information about the characteristics of soil types and bedrock of the NEF site and its vicinity and design-basis earthquake magnitudes and return periods. The WCS site in Texas and the former proposed Atomic Vapor Laser Isotope Separation (AVLIS) site, located in Section 33, have both been thoroughly studied in recent years in preparation for construction of other facilities. A review of those documents and related materials provides a significant description of geological conditions pertinent to the NEF site. In addition, LES performed field confirmation, where necessary, in order to clarify any questions about regional or site-specific conditions.

1.3 Site Description

The NEF site is located in New Mexico immediately west of the Texas border about 48 km (30 mi) from the extreme southeast corner of the state and about 96 km (80 mi) east of the Pecos River. The site is contained in the Eunice NE, Texas-New Mexico USGS topographic quadrangle (USGS, 1979). This location is near the boundary between the Pecos Plains Section to the west; and the Southern High Plains Section of the Great Plains province to the east. The boundary between the two sections is the Mescalero Escarpment, locally referred to as Mescalero Ridge.

NEF site elevations range between +1033 and +1045 m (+3390 and +3430 ft) (msl). The finished site grade is about +1041 m (+3415 ft) msl.

Surface exposures of geologic units at the site include surficial eolian deposits and Tertiary-aged alluvium. These overlie Triassic red-bed clay which overlies sedimentary rock. The principal underlying geologic structure is the Central Basin Platform which divides the Permian Basin into the Midland and Delaware sub-basins.

1.3.5.1 Characteristics Of Soil Types And Bedrock

The dominant subsurface structural feature of this region is the Permian Basin. This 250 million-year-old feature is the source of the Region's prolific oil and gas reserves.

The NEF site is located within the Central Permian Basin Platform area, where the top of the Permian deposits are approximately 434 to 480 m (1,425 to 1,575 ft) below ground surface. Overlying the Permian are the sedimentary rocks of the Triassic Age Dockum Group.

Soil development in the region is generally limited due to its semi-arid climate. The site has a minor thickness of soil (generally less than 0.4 m (1.4 ft)) developed from subaerial weathering. A small deposit of active dune sand is present at the southwest corner of the site. The U. S. Department of Agriculture soil survey for Lea County, New Mexico (USDA, 1974) categorizes site soils as hummocky loamy (silty) fine sand with moderately rapid permeability and slow runoff, well-drained non-calcareous loose sand, active dune sand and dune-associated sands.

Recent deposits are primarily dune sands derived from Permian and Triassic rocks of the Permian Basin. These Mescalero (dune) Sands cover over 80% of Lea County and are generally described as fine to medium-grained and reddish brown in color. The USDA Soil Survey of Lea County identifies the dune sands at the site as either the Brownsfield-Springer Association of reddish brown fine to loamy fine sands; or the Gomez series of brown to yellowish brown loamy fine sand (USDA, 1974).

1.3.5.2 Earthquake Magnitudes And Return Periods

The majority of earthquakes in the United States are located in the tectonically active western portion of the country. However, areas within New Mexico and the southwestern United States also experience earthquakes, although at a lower rate and at lower intensities. Earthquakes in the region around the NEF site include isolated and small clusters of low to moderate size events toward the Rio Grande Valley of New Mexico and in Texas, southeast of the NEF site.

The largest earthquake within 322 km (200 mi) of the NEF is the August 16, 1931 earthquake located near Valentine, Texas. This earthquake has an estimated magnitude of 6.0 to 6.4 and produced a maximum epicentral intensity of VIII on the Modified Mercalli Intensity (MMI) Scale. The intensity observed at the NEF site is IV on the MMI scale.

A site-specific probabilistic seismic hazard analysis was performed for the NEF site using the seismic source zone geometries and earthquake recurrence models. The modeling included attenuation models suited for the regional and local seismic wave transmission characteristics.

Total seismic ground motion hazard to a site results from summation of ground motion effects from all distant and local seismically active areas. The 250-year and 475-year return period peak horizontal ground accelerations are estimated at 0.024 g and 0.036 g, respectively. The 10,000 year return period peak horizontal ground acceleration is estimated at 0.15 g. The associated peak vertical ground motion is estimated at 0.10 g.

1.3.5.3 Other Geologic Hazards

There are no other known geologic hazards that would adversely impact the NEF site.

1.4 References

Edition of Codes, Standards, NRC Documents, etc that are not listed below are given in ISAS Table 3.0-1.

AMS, 1996. Glossary of Weather and Climate, With Related Oceanic and Hydrologic Terms, American Meteorological Society, 1996.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 61, Licensing Requirements for Land Disposal of Radioactive Waste, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 70.22, Contents of Applications, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.25, Financial Assurance and Recordkeeping for Decommissioning, 2003.

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1.4 References

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1.5 Chapter 1 Tables

Table 1.1-1 Estimated Annual Gaseous Effluent

Area	Quantity (yr^{-1})	Discharge Rate m^3/yr (SCF/yr) (STP^3)
GEVS (Note 1)	NA	2.3×10^8 @ Standard Temperature and Pressure (STP) (8.09×10^9)
HVAC Systems		
Radiological Areas	NA	1.5×10^9 (5.17×10^{10})
Non-Radiological Areas	N/A	1.0×10^9 (3.54×10^{10})
Total Gaseous HVAC Discharge	NA	2.47×10^9 (8.71×10^{10})
Constituents:		
Helium	440 m^3 @ (STP) (15,536 ft^3)	NA
Nitrogen	52 m^3 @ (STP) (1,836 ft^3)	NA
Ethanol	40 L (10.6 gal)	NA
Laboratory Compounds	Traces (HF) (NA)	NA
Argon	190 m^3 (6,709 ft^3)	NA
Hydrogen Fluoride	< 1.0 kg (< 2.2 lb)	NA
Uranium	< 10 g (< 0.0221 lb)	NA
Methylene Chloride	610 L (161 gal)	NA

N/A – Not applicable

Note 1. This includes the monitored gaseous discharges from Pumped Extract GEVS, CRDB GEVS, and the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System.

1.5 Chapter 1 Tables

Table 1.1-2 Estimated Annual Radiological and Mixed Wastes[1]

Waste Type	Radiological Waste		Mixed Waste[2]	
	Total Mass kg (lb)	Uranium Content kg (lb)	Total Mass kg (lb)	Uranium Content kg (lb)
Activated Carbon	300 (662)	25 (55)	-	-
Activated Alumina	2160 (4763)	2.2 (4.9)	-	-
PFPE Oil Recovery Sludge	20 (44)	5 (11)	-	-
Liquid Waste Treatment Sludge	400 (882)	57 (126)	-	-
Activated Sodium Fluoride[3]	-	-	-	-
Assorted Materials (paper, packing, clothing, wipes, etc.)	2100 (4,631)	30 (66)	-	-
Ventilation Filters	61,464 (135,506)	5.5 (12)		-
Non-Metallic Components	5000 (11,025)	Trace[4]	-	-
Miscellaneous Mixed Wastes (organic compounds)[5]			50 (110)	2 (4.4)
Combustible Waste	3,500 (7,718)	Trace[4]	-	-
Scrap Metal	12,000 (26,460)	Trace[4]	-	-

Table 1.1-3 Estimated Annual Liquid Effluent

Summation of Liquid Effluents (excluding utilities)	Gal/Day	Gal/Yr	Liters/sec
Floor Washings, Misc. condensates, and Lab effluent	17	6,112	0.0
Degreaser Water	3	980	0.0
Citric Acid	2	719	0.0
Hand Wash and Shower Water	1,520	554,820	0.1
Total Liquid Effluents	1,542	562,631	0.1

2 Valves were based on initial licensed facility design. More accurate forecasts of waste generation volumes will be based on operating history along with process knowledge.

2 A mixed waste is a low-level radioactive containing listed or characteristic of hazardous wastes as specified in 40 CFR 261, Subparts C and D.

3 No sodium fluoride (NaF) wastes are produced on an annual basis. The contingency dump system NaF traps are not expected to saturate over the life of the plant.

4 Trace is defined as not detectable above naturally occurring background concentrations.

5 Representative organic compounds consist of acetone, toluene, ethanol, and petroleum ether.

1.5 Chapter 1 Tables

Table 1.1-4 Estimated Annual Non-Radiological Wastes

Waste	Annual Quantity
Spent Blasting Sand*	125 kg (275 lbs)
Miscellaneous Combustible Waste*	9000 kg (19,800 lbs)
Cutting Machine Oils	45 L (11.9 gal)
Spent Degreasing Water (from ME&I workshop)	1 m^3 (264 gal)
Spent Demineralizer Water (from ME&I workshop)	200 L (53 gal)
Empty Spray Paint Cans*	20 ea
Empty Cutting Oil Cans	20 ea
Empty Propane Gas Cylinders*	5 ea
Acetone*	27 L (7.1 gal)
Toluene*	2 L (0.5 gal)
Degreaser Solvent SS25*	2.4 L (0.6 gal)
Petroleum Ether*	10 L (2.6 gal)
Diatomaceous Earth*	10 kg (22 lbs)
Miscellaneous Scrap metal	2,800 kg (6.147 lbs)
Motor Oils (For internal combustion. engines)	3,400 L (895 gal)
Oil Filters	250 ea
Air Filters (vehicles)	50 ea
Air Filters (building ventilation)	160,652 kg (354,200 lb)
Hydrocarbon Sludge*	10 kg (22 lbs)
Methylene Chloride*	1850 L (487 gal)

* Hazardous waste as defined in Title 40, Code of Federal Regulations, Part 261, Identification and listing of hazardous waste, 2003. (in part or whole)

1.5 Chapter 1 Tables

Table 1.1-5 Annual Hazardous Construction Wastes

Waste Type	Annual Quantity
Paint, Solvents, Thinners, Organics	1,134 L (3,000 gal)
Petroleum Products – Oils, Lubricants	1,134 L (3,000 gal)
Sulfuric Acid (Batteries)	380 L (100 gal)
Adhesives, Resins, Sealers, Caulking	910 kg (2,000 lbs)
Lead (Batteries)	91 kg (200 lbs)
Pesticide	380 L (100 gal)

Table 1.2-1 Type, Quantity and Form of Licensed Material

Source and/or Special Nuclear Material	Physical and Chemical Form	Maximum Amount to be Possessed at Any One Time
Uranium (natural and depleted) and daughter products	Physical: Solid, Liquid and Gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	See Material License Condition 8A for limit.
Uranium enriched in isotope ^{235}U up to the LES license limit (See Material License Condition 6B for limit)	Physical: Solid, Liquid, and Gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	See Material License Condition 8B for limit.
^{99}Tc, transuranic isotopes and other contamination	Any	Amount that exists as contamination as a consequence of the historical feed of recycled uranium at other facilities(1)

(1) To minimize potential sources of contamination of UF_6, such as ^{99}Tc, LES will require UF_6 suppliers to provide Commercial Natural UF_6 in accordance with ASTM C 787, "Standard Specification for Uranium Hexafluoride for Enrichment." In addition, cylinder suppliers will be required to preclude use of cylinders that, in the past, have contained reprocessed UF_6, unless they have been decontaminated. Periodic audits of suppliers will be performed to provide assurance that these requirements are satisfied.

1.6 Chapter 1 Figures

1.6 Chapter 1 Figures

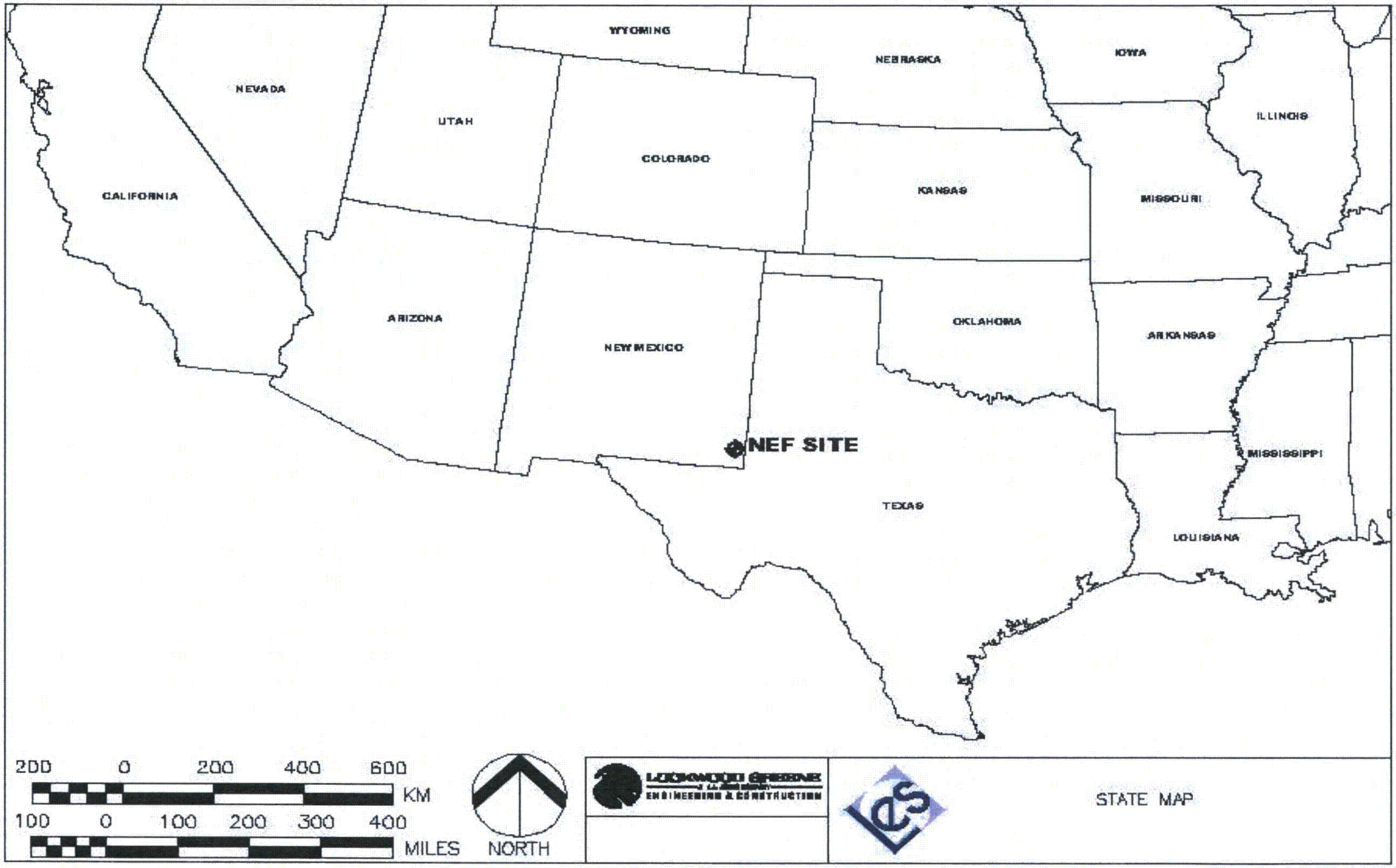


Figure 1.1-1 State Map

1.6 Chapter 1 Figures

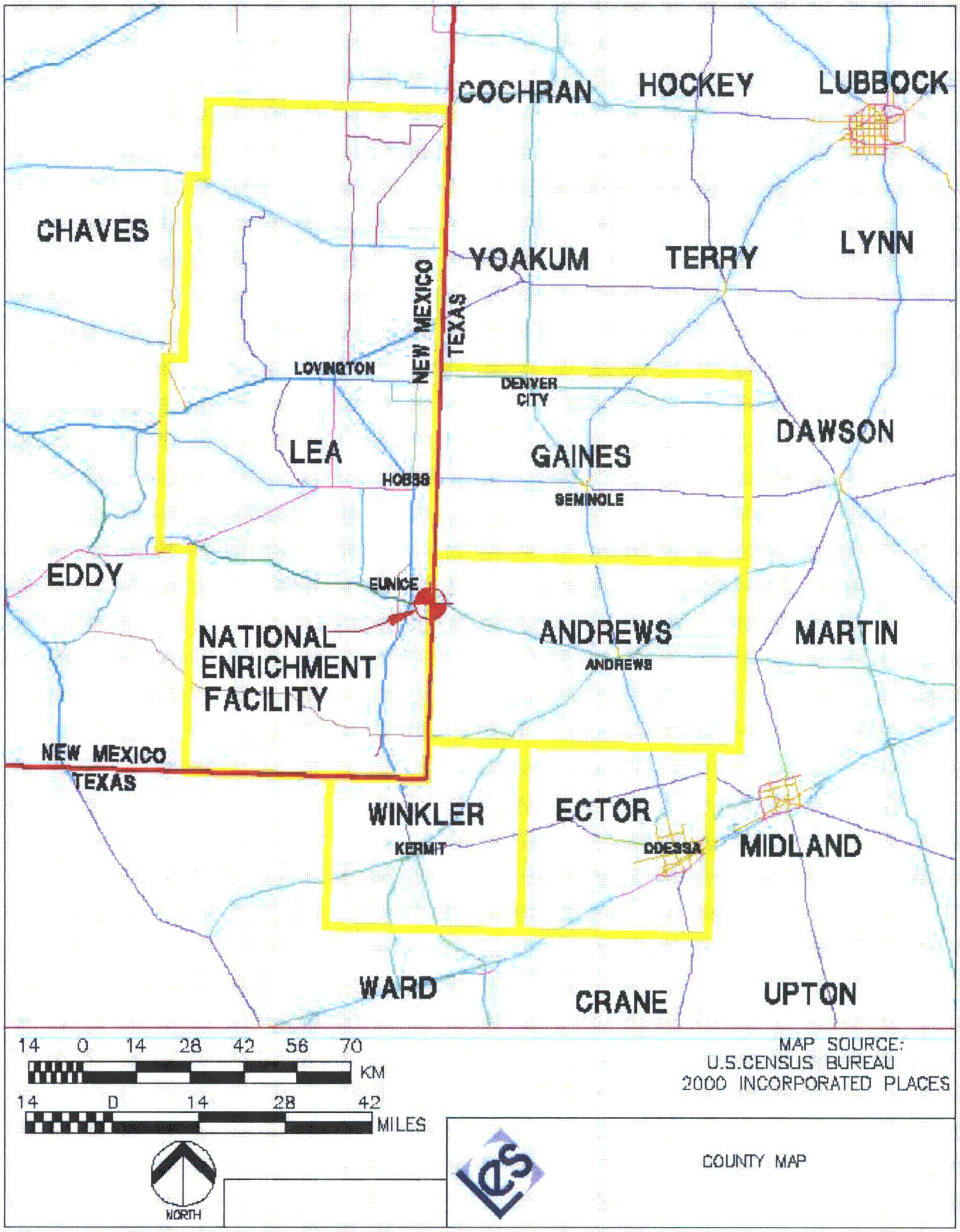


Figure 1.1-2 County Map

1.6 Chapter 1 Figures

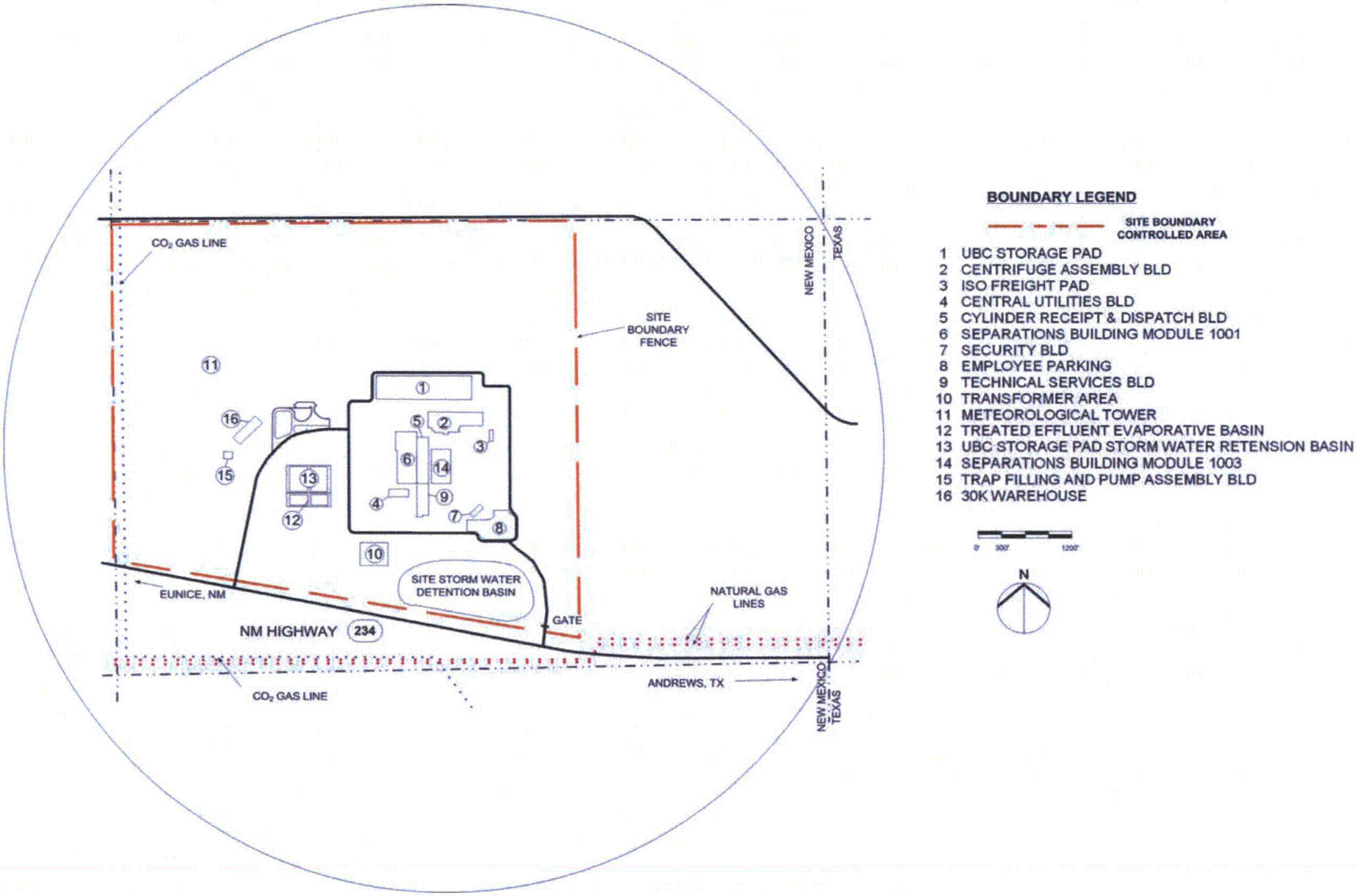


Figure 1.1-3 Plot Plan (1 Mile Radius)

1.6 Chapter 1 Figures

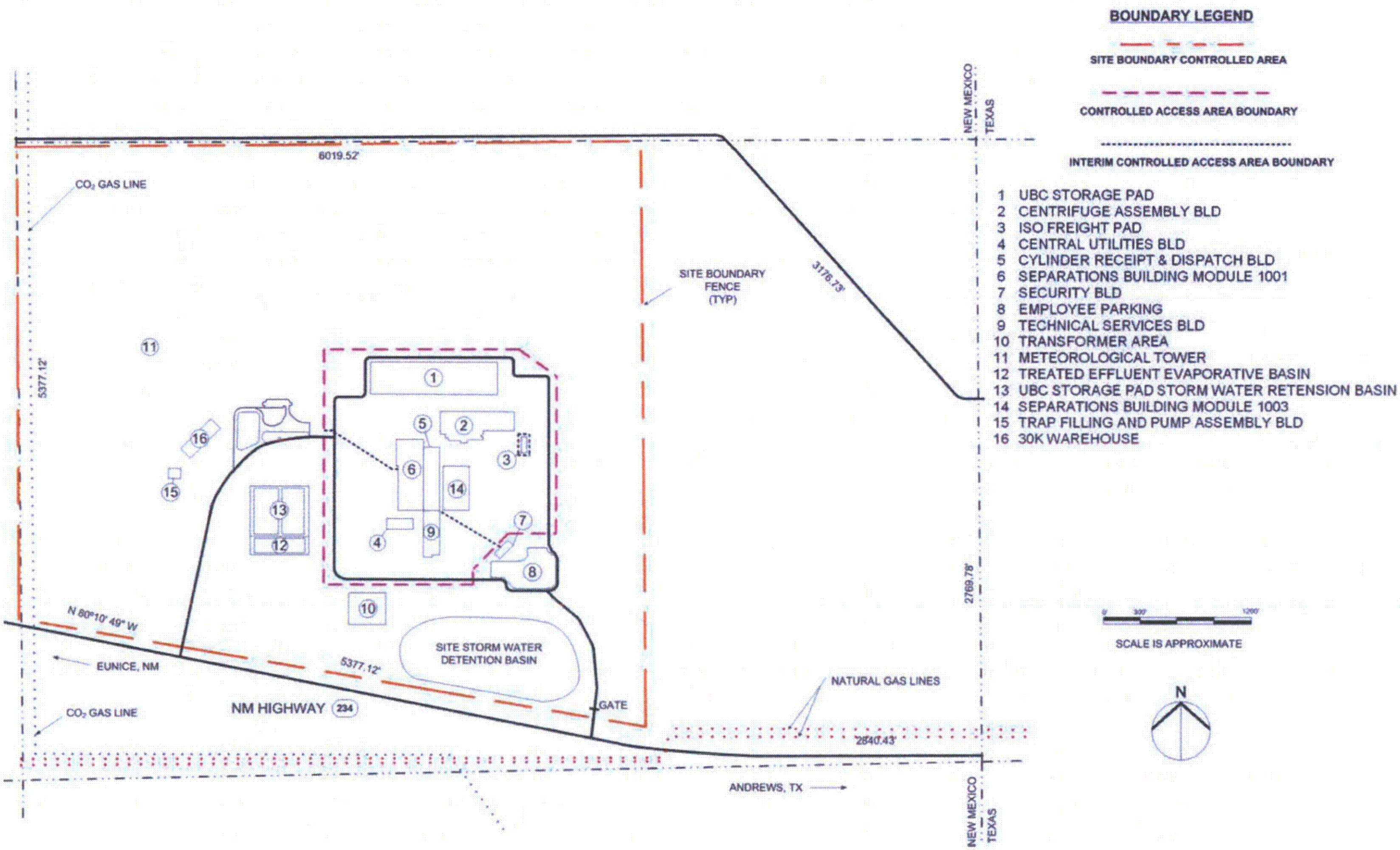


Figure 1.1-4 Facility Layout (Site Plan) with Site Boundary and Controlled Access Area Boundary

1.6 Chapter 1 Figures

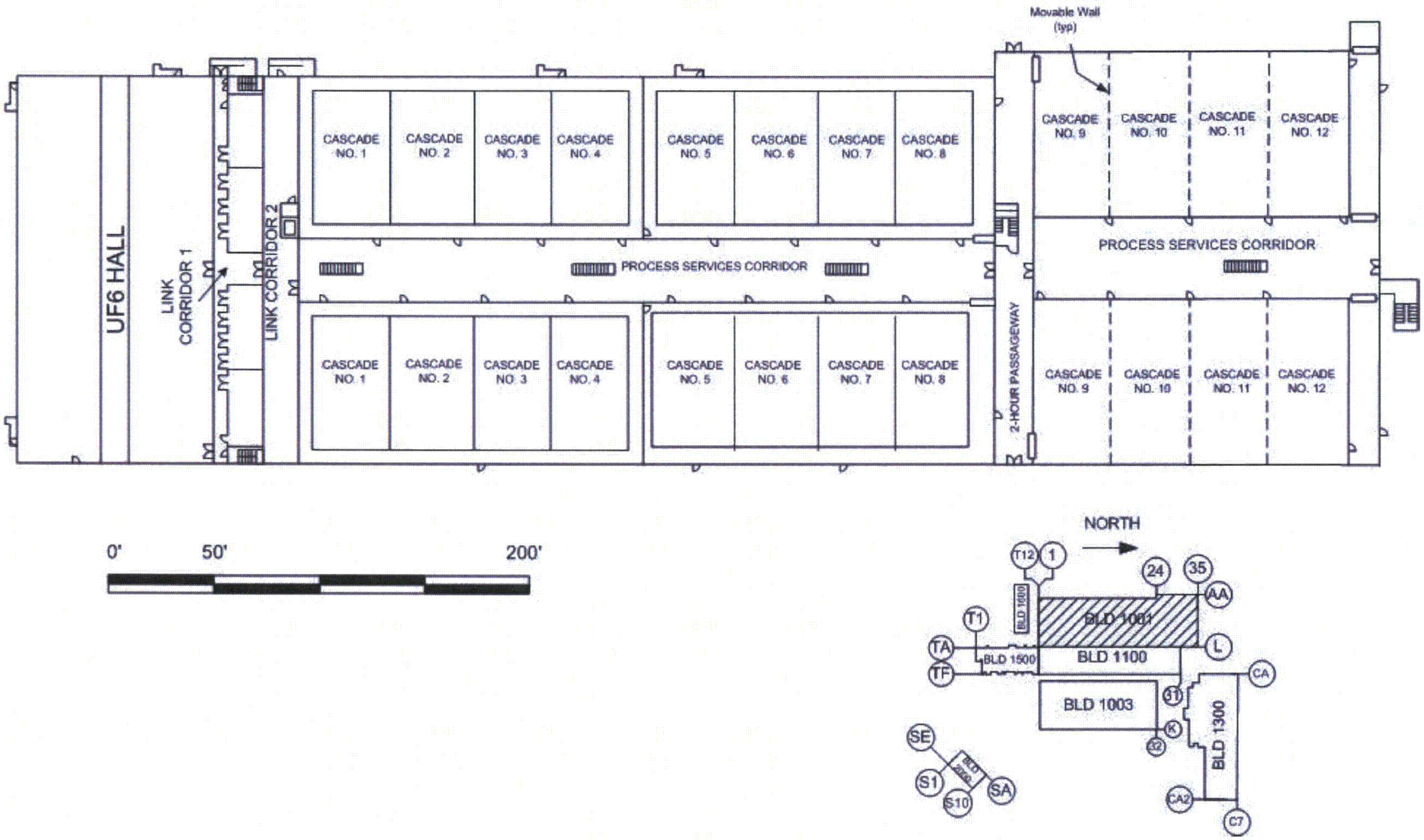


Figure 1.1-5 Separations Building Module 1001 First Floor

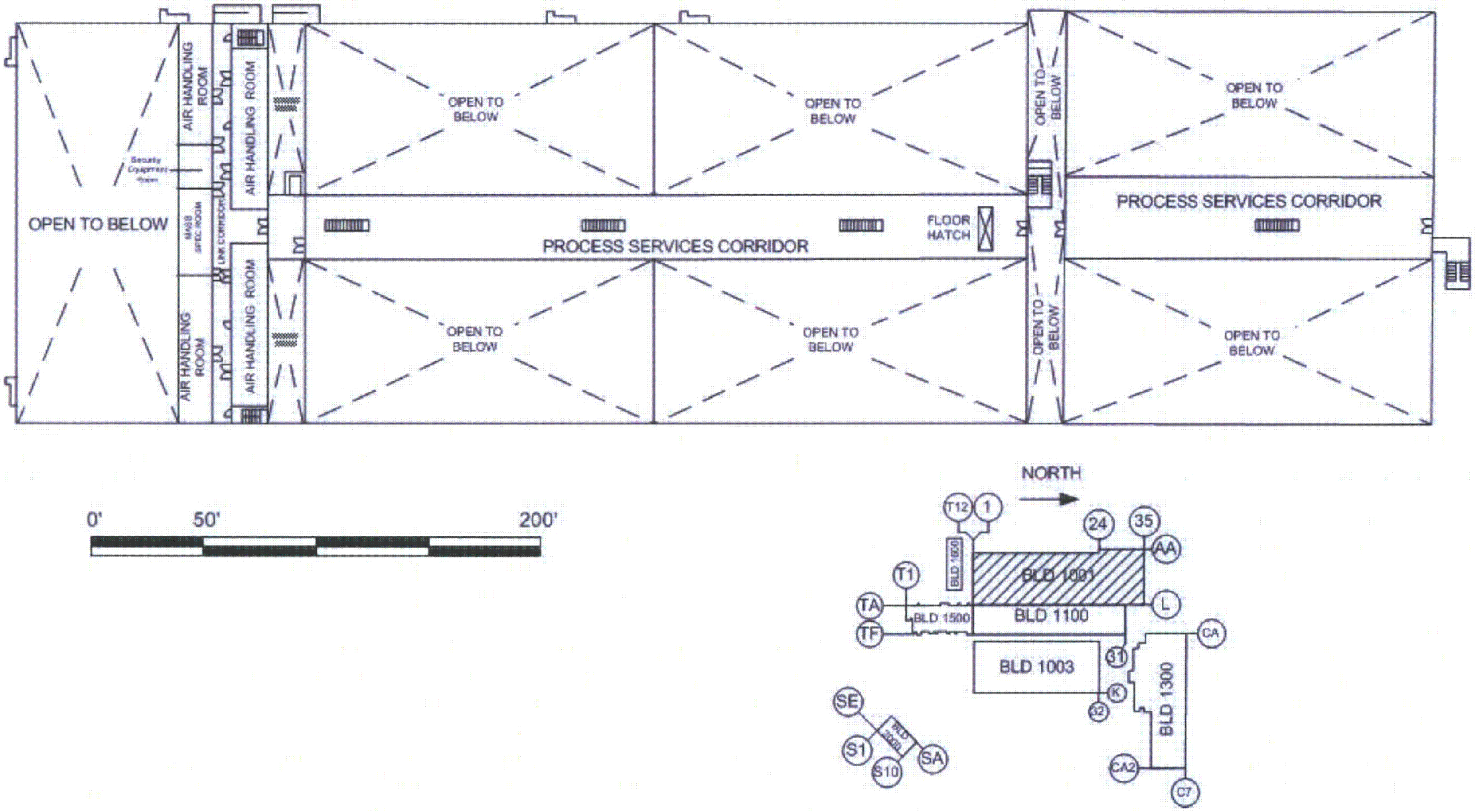


Figure 1.1-6 Separations Building Module 1001 Second Floor

1.6 Chapter 1 Figures

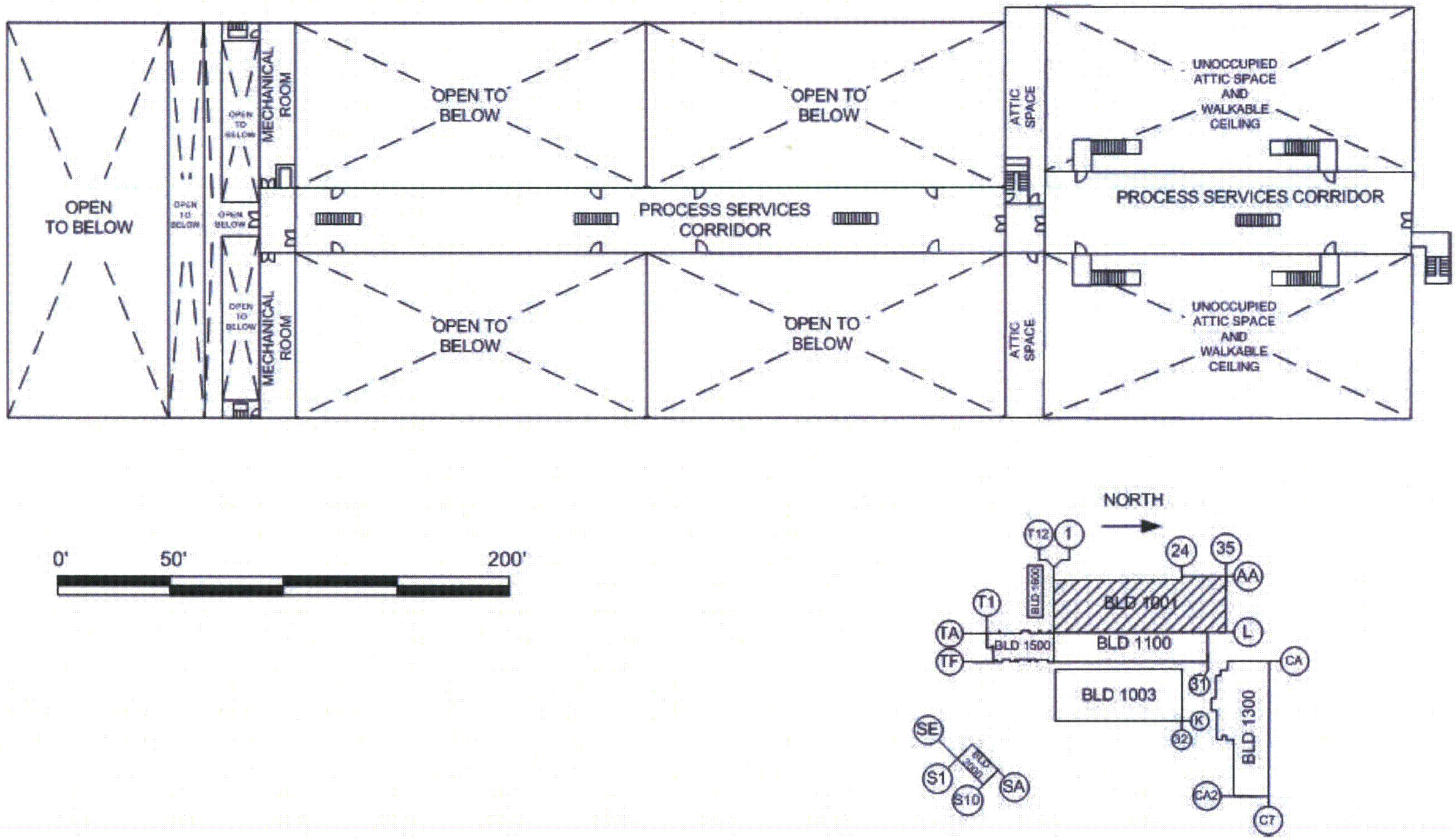
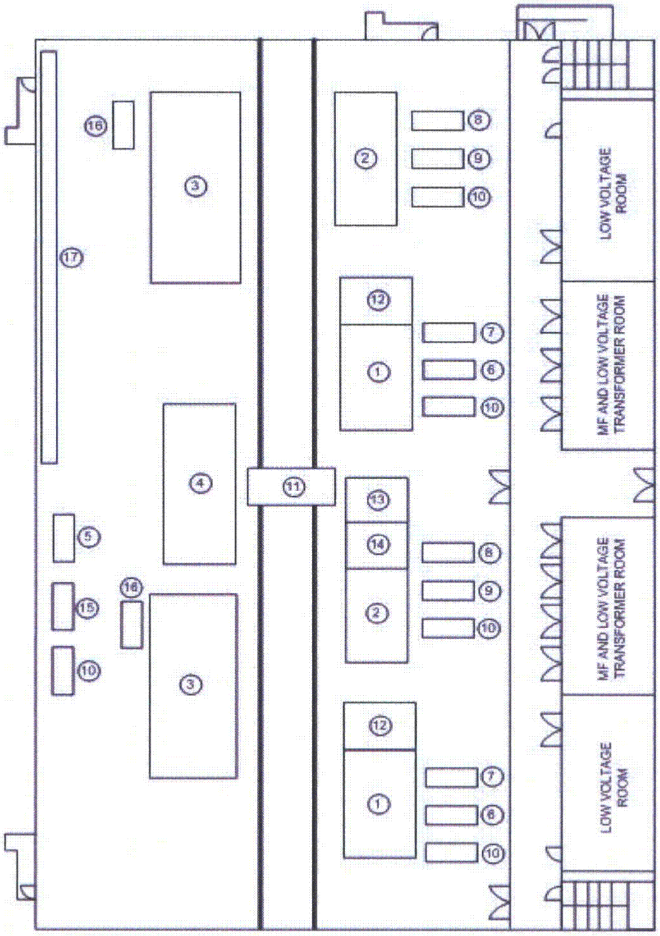
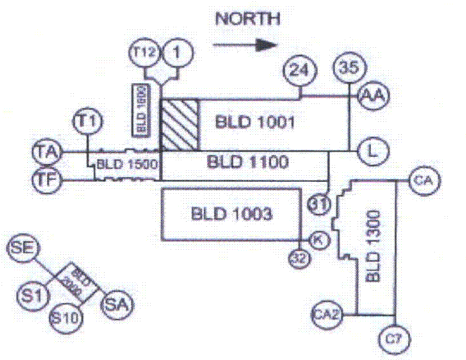


Figure 1.1-7 Separations Building Module 1001 Third Floor





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

Figure 1.1-8 Separations Building Module 1001 UF_6 Handling Area Equipment Location

1.6 Chapter 1 Figures

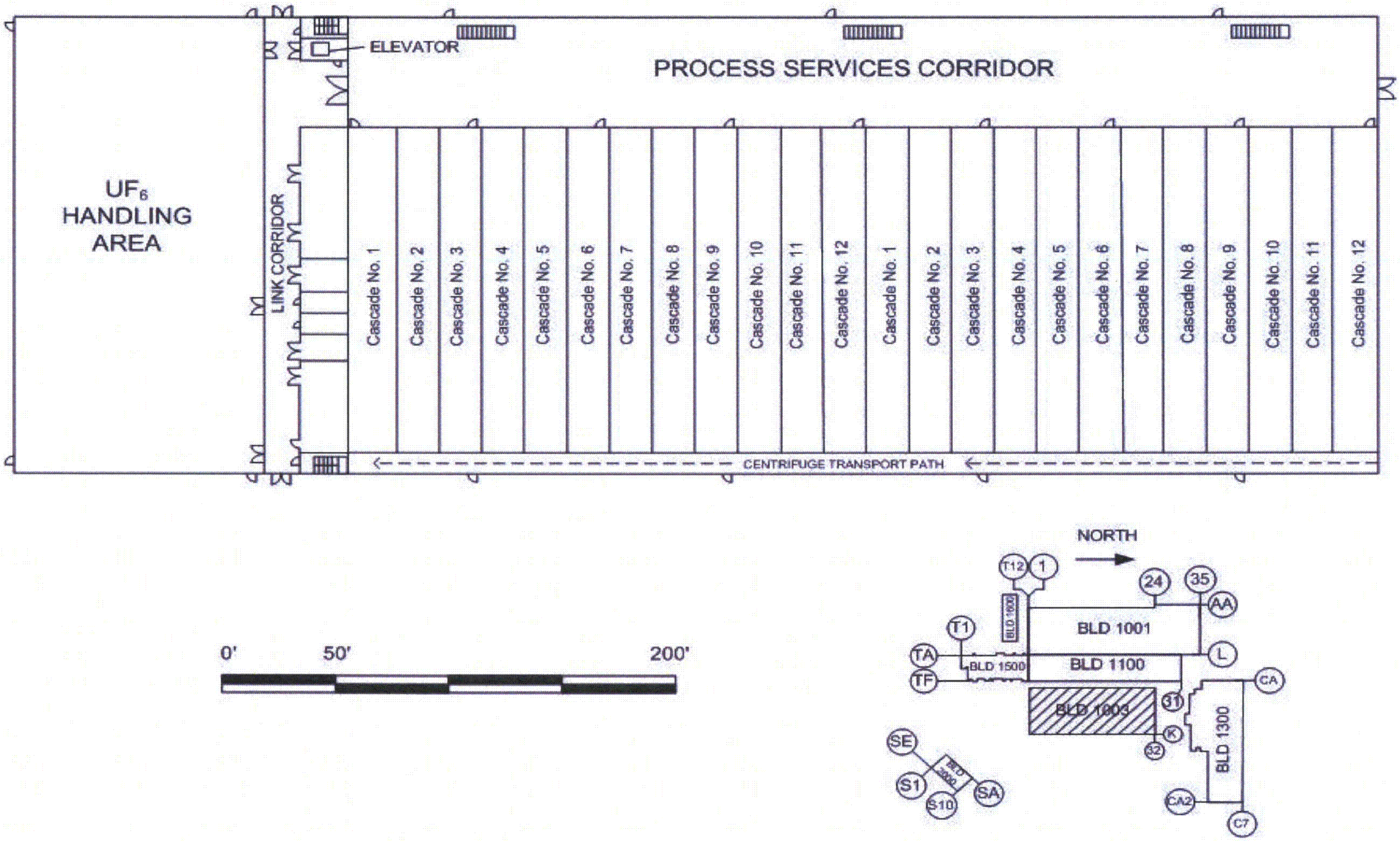


Figure 1.1-9 Separations Building Module 1003 First Floor

1.6 Chapter 1 Figures

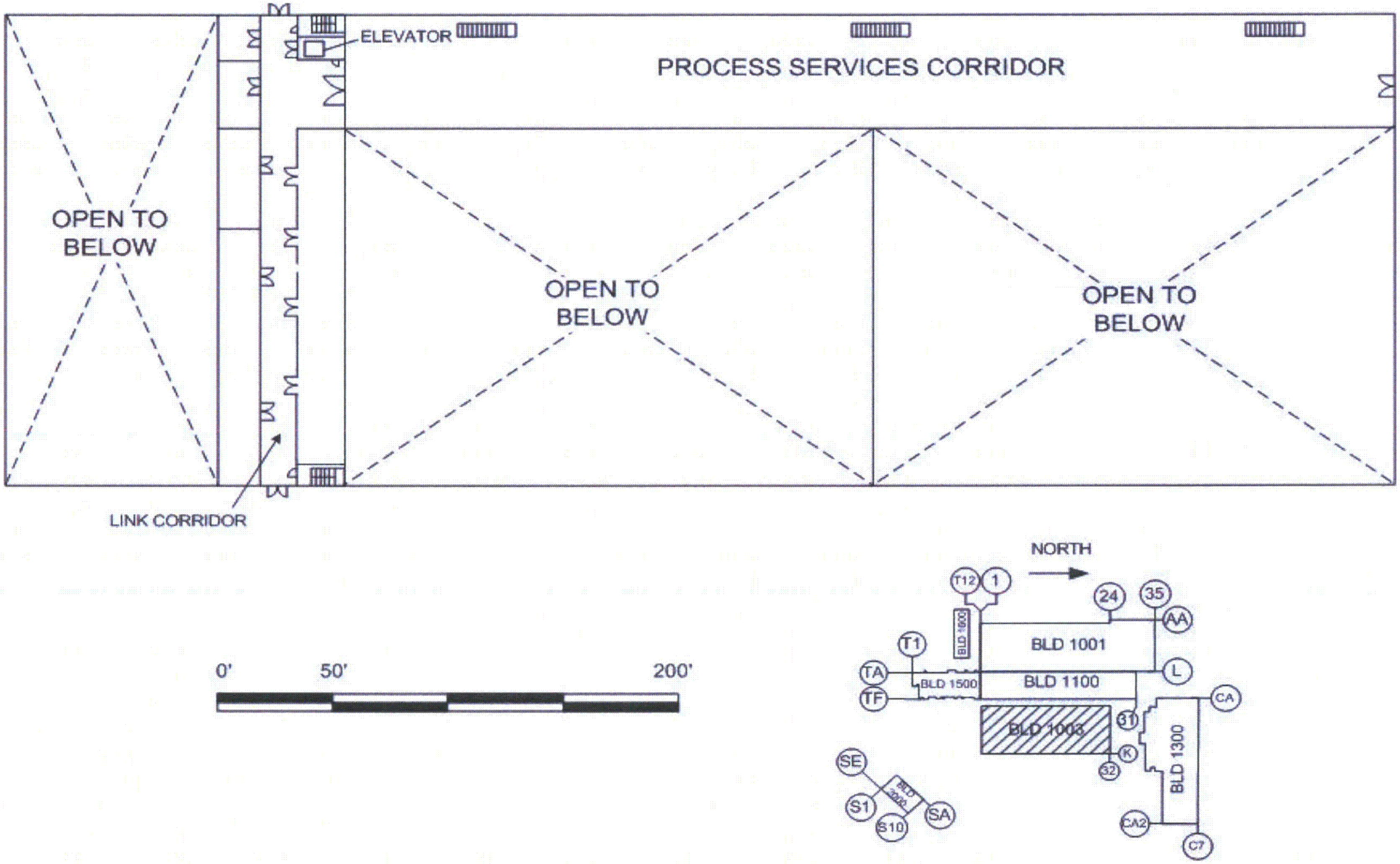


Figure 1.1-10 Separations Building Module 1003 Second Floor

1.6 Chapter 1 Figures

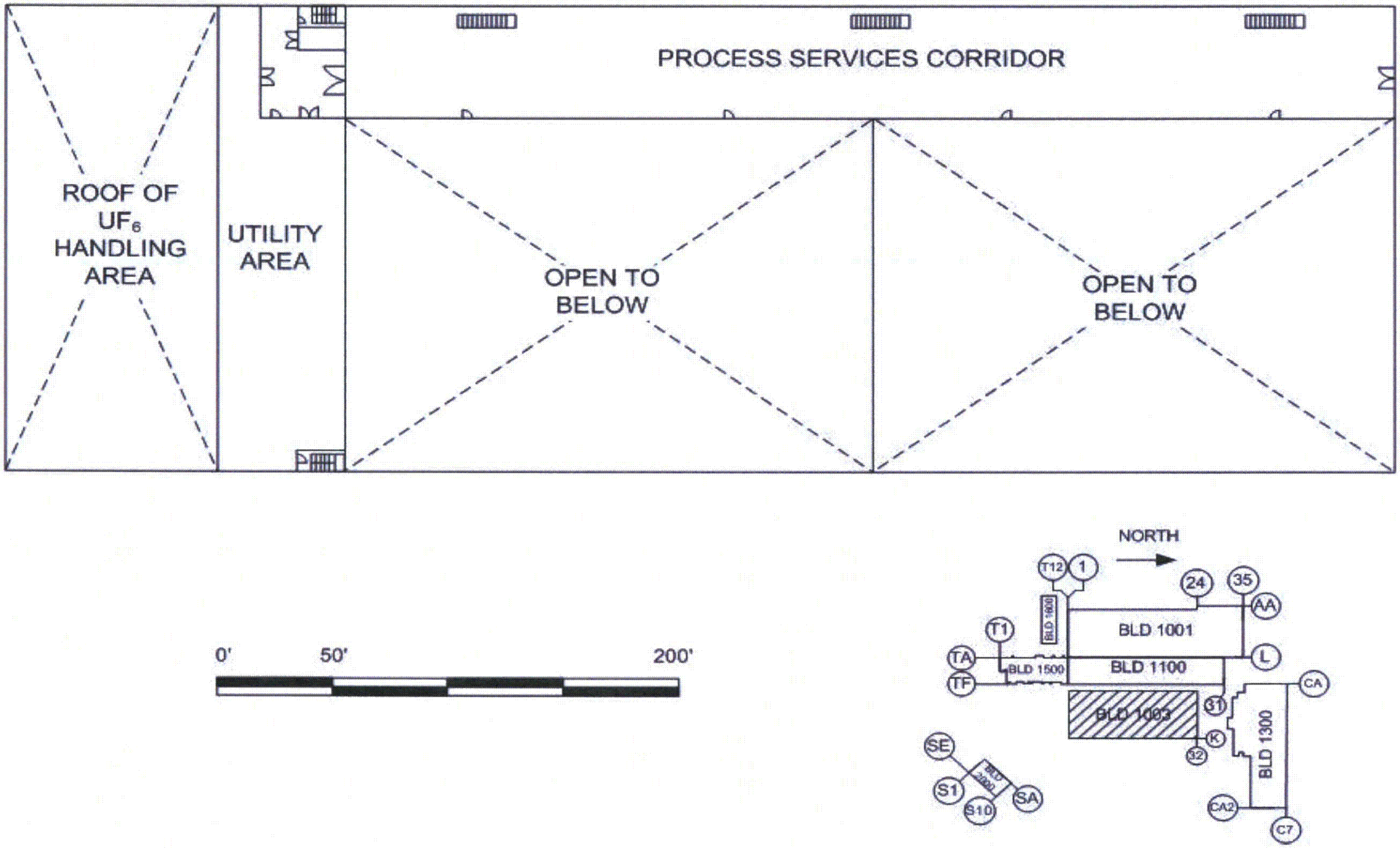
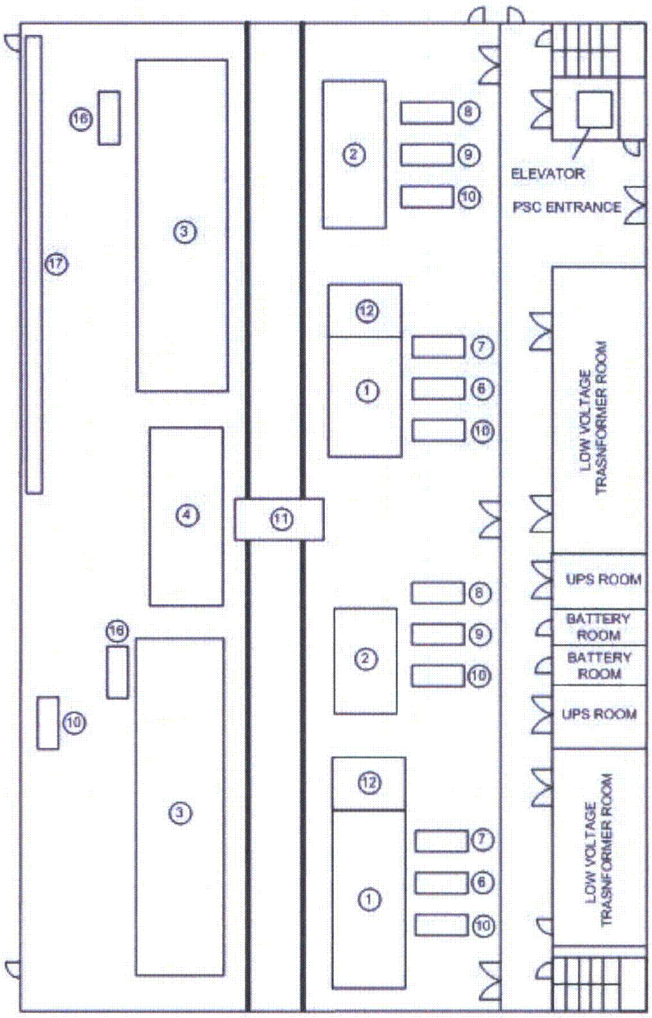


Figure 1.1-11 Separations Building Module 1003 Third Floor

1.6 Chapter 1 Figures



1 SOLID FEED STATIONS
2 PRODUCT STATIONS
3 TAILS STATIONS
4 AUTOCLAVES
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
16 TAILS VACUUM PUMP TRAP SET
17 PUMPED EXTRACT GEVS

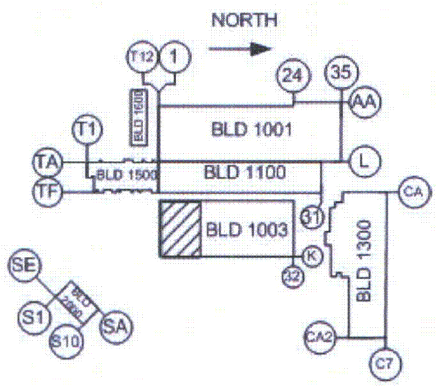


Figure 1.1-12 Separations Building Module 1003 UF_6 Handling Area Equipment Location

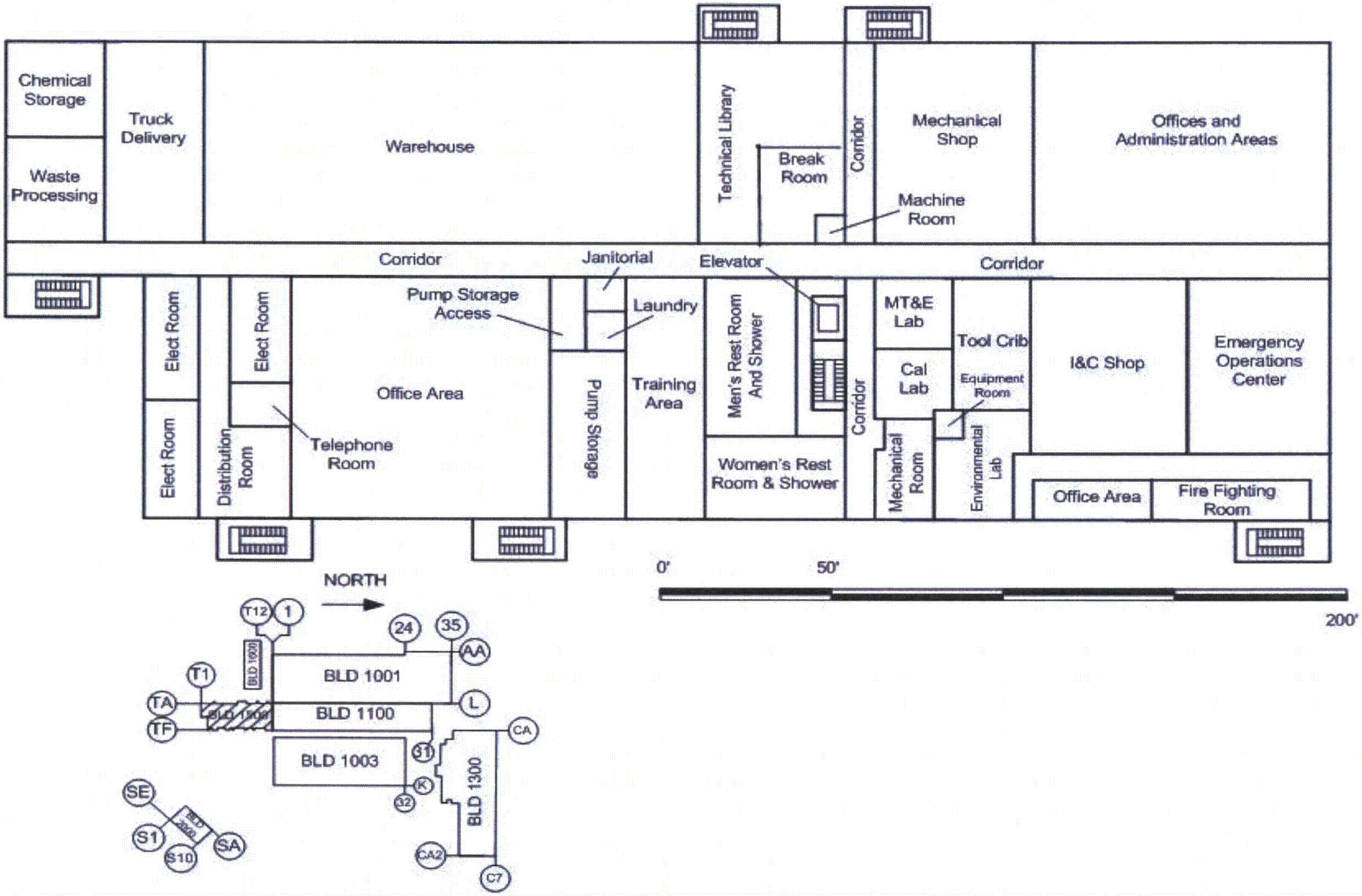


Figure 1.1-13 Technical Services Building First Floor

1.6 Chapter 1 Figures

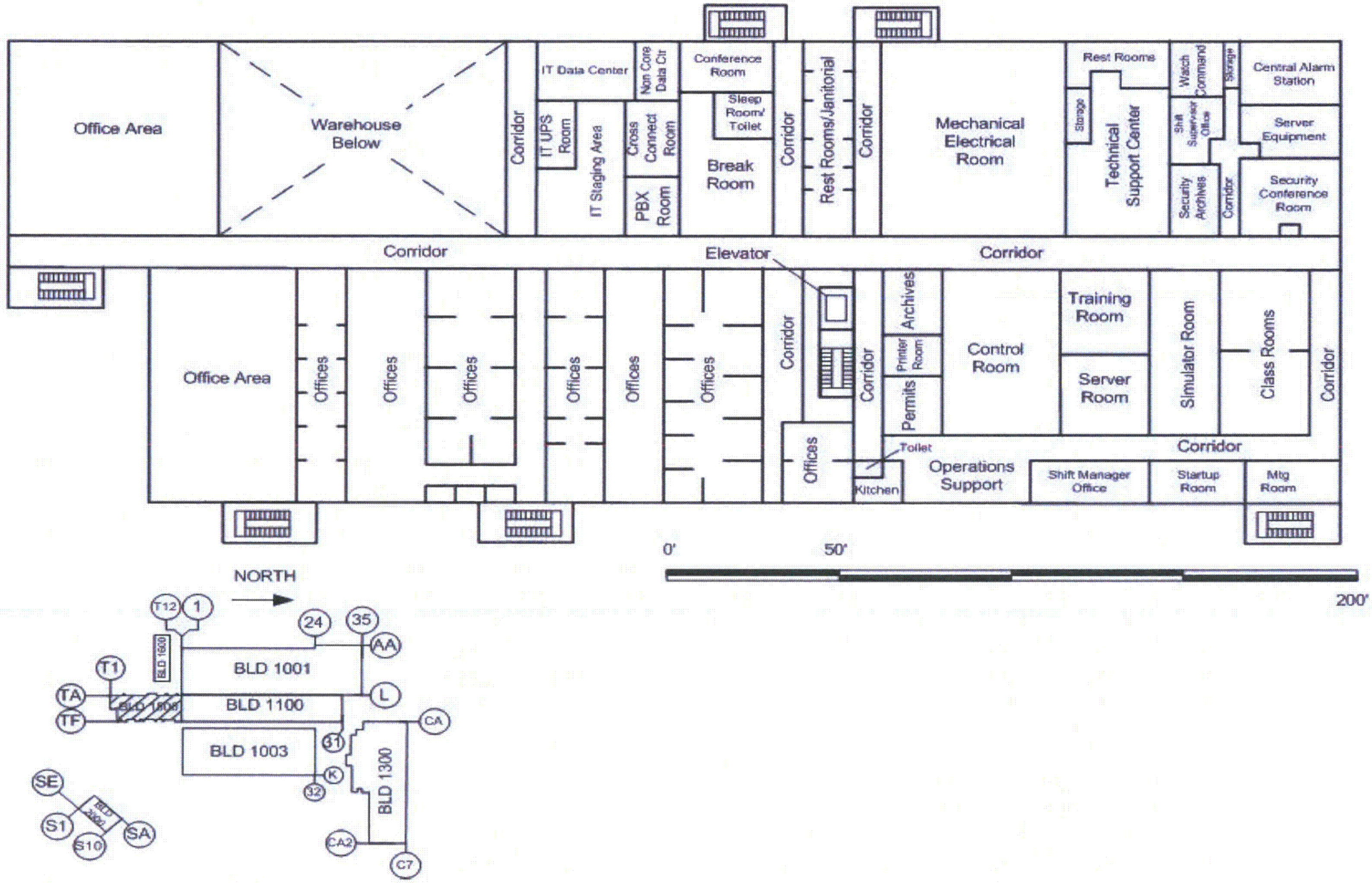


Figure 1.1-14 Technical Services Building Second Floor

1.6 Chapter 1 Figures

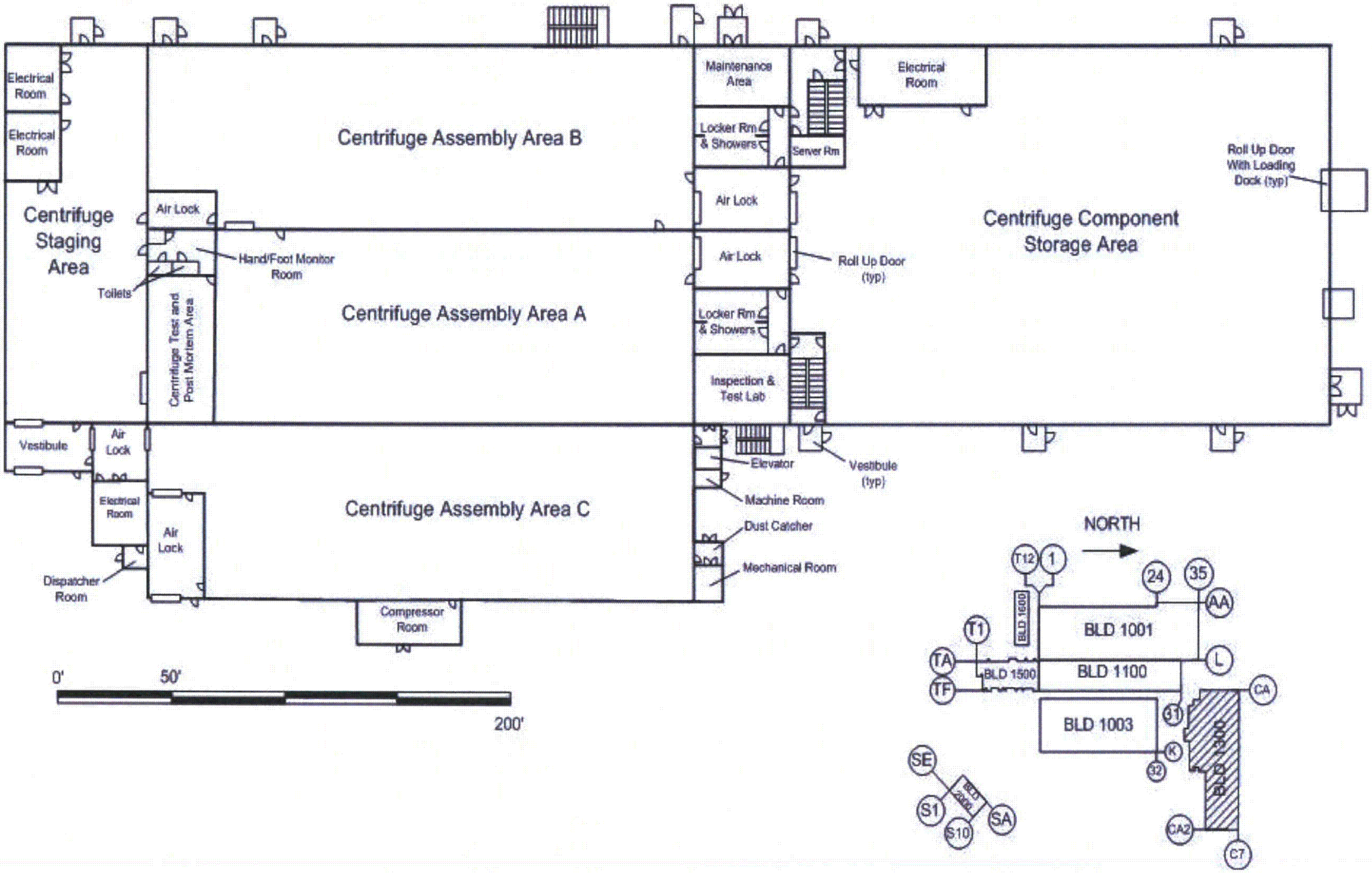


Figure 1.1-15 Centrifuge Assembly Building First Floor

1.6 Chapter 1 Figures

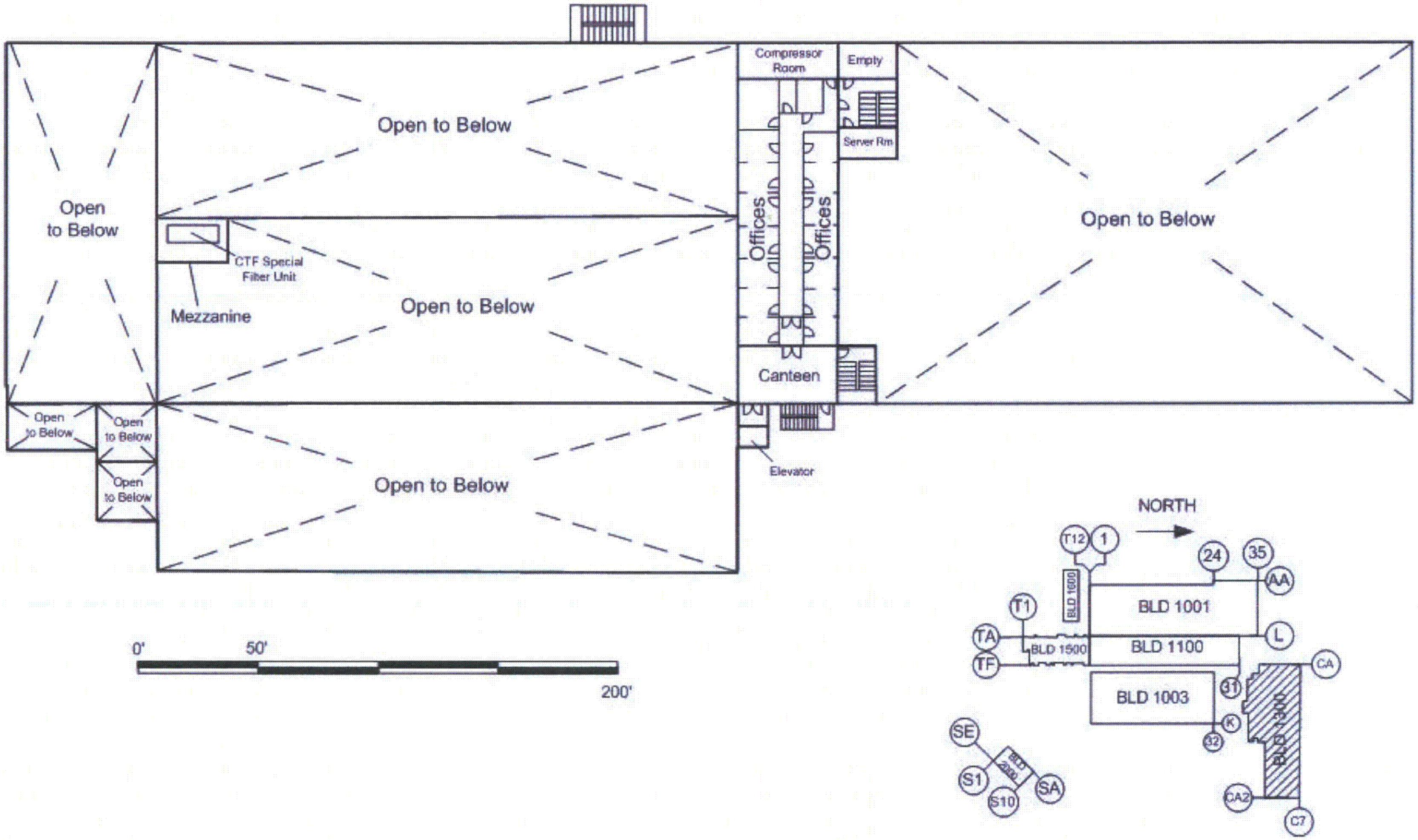
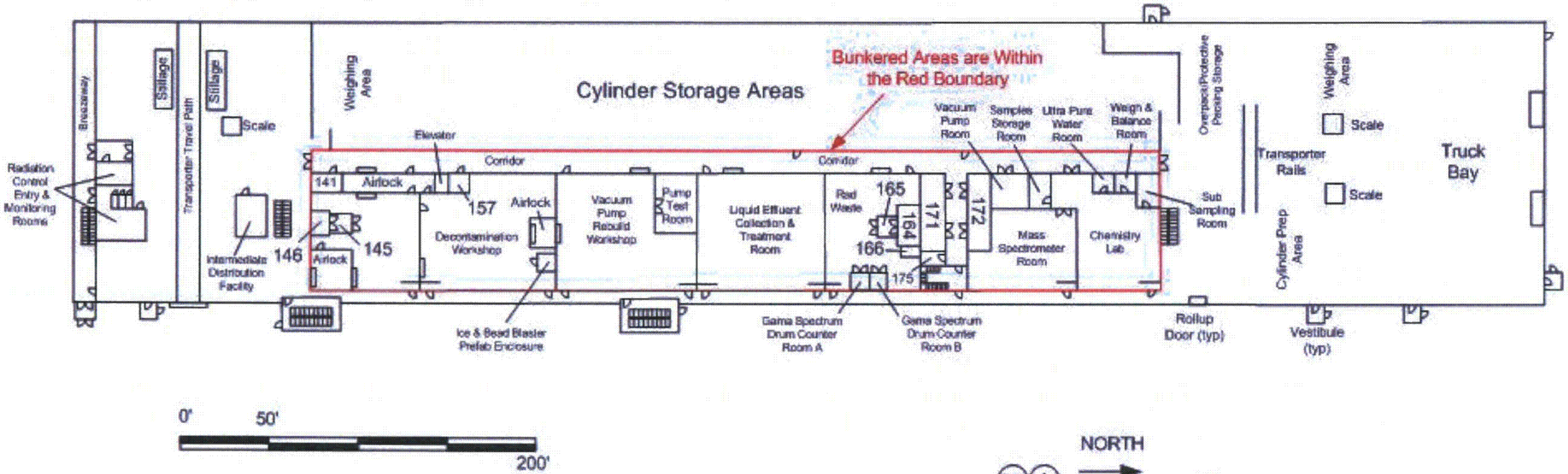


Figure 1.1-16 Centrifuge Assembly Building Second Floor

1.6 Chapter 1 Figures



Room No.	Description
141	Personnel Decontamination Room
145	Cold Trap and Filter Unit Enclosure
146	Trap Emptying and Drum Tipper 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

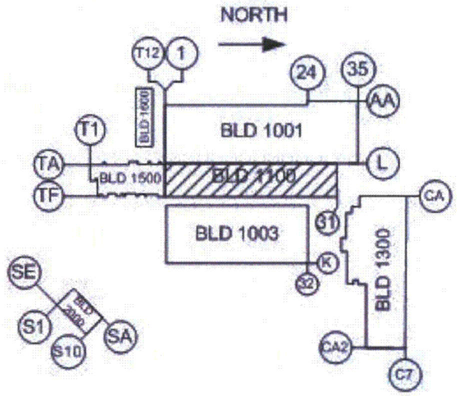


Figure 1.1-17 Cylinder Receipt and Dispatch Building First Floor

1.6 Chapter 1 Figures

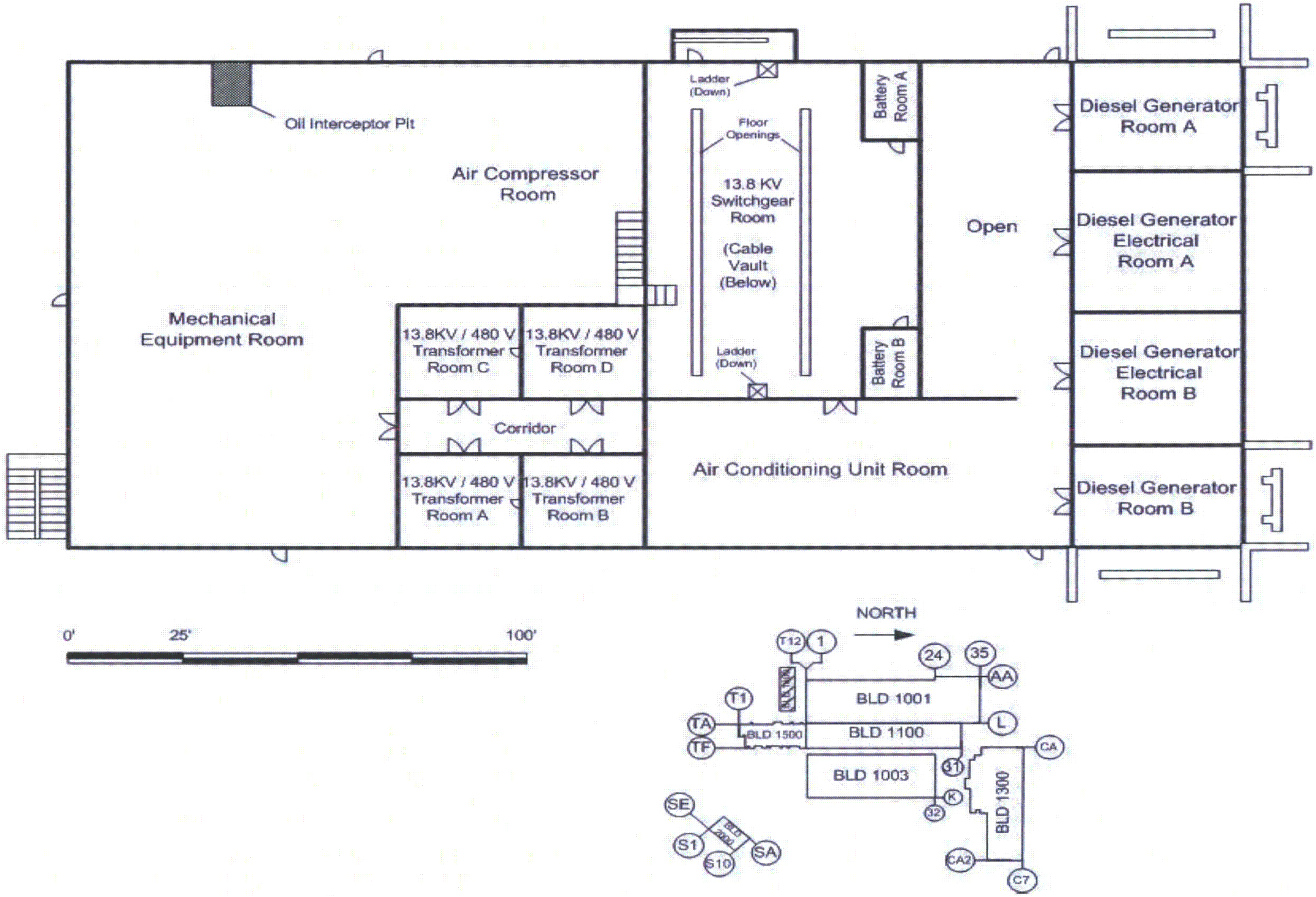


Figure 1.1-18 Central Utilities Building First Floor

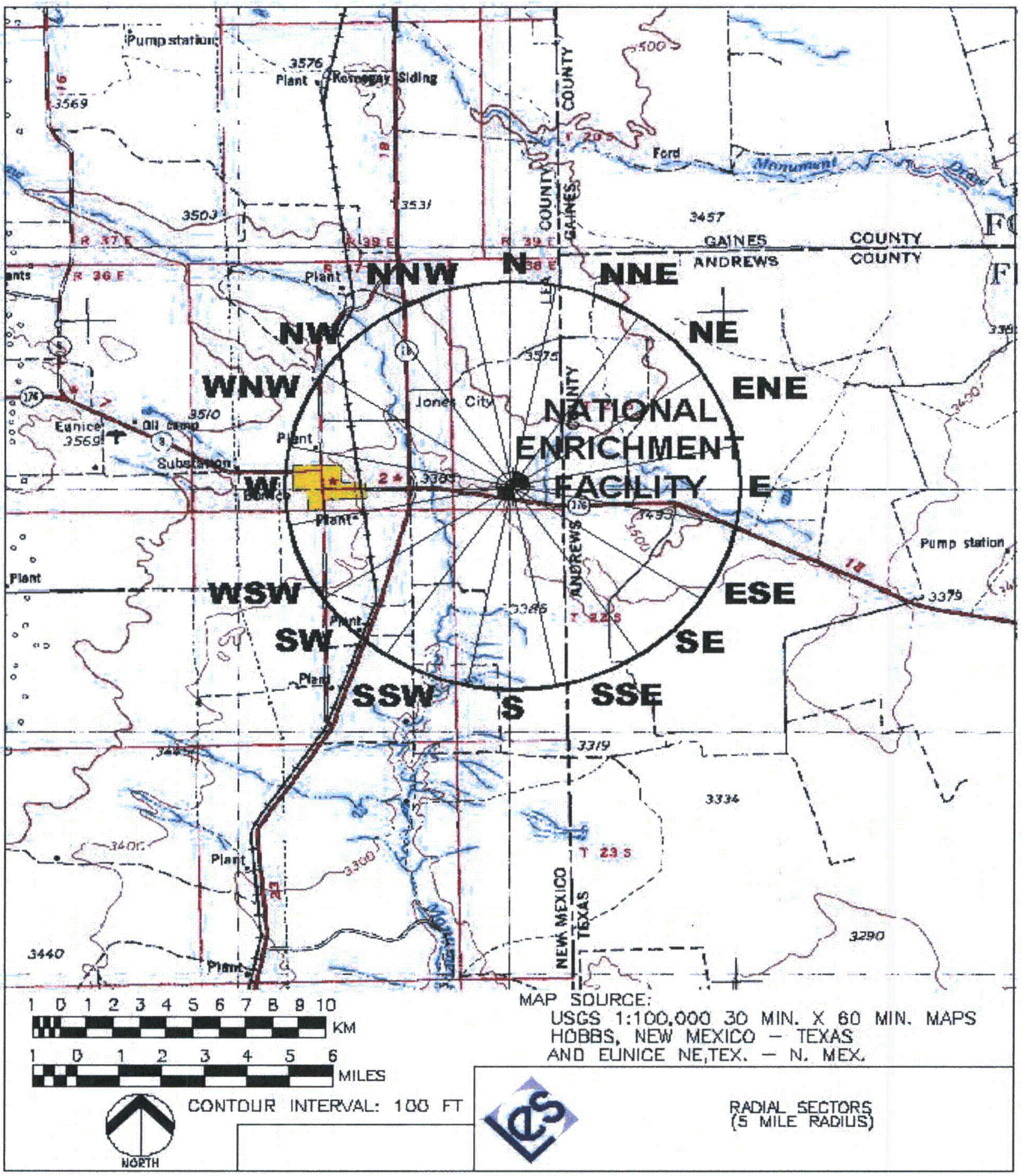


Figure 1.3-1 Radial Sectors (5 Mile Radius) Organization and Administration

2.0 Organization and Administration

This chapter describes the management system and administrative procedures for the effective implementation of Health, Safety, and Environmental (HS&E) functions at the Louisiana Energy Services (LES) enrichment facility. The chapter presents the organizations responsible for managing the design, construction, operation, and decommissioning of the facility. The key management and supervisory positions and functions are described including the personnel qualifications for each key position at the facility.

The LES policy is to maintain a safe work place for its employees and to assure operational compliance within the terms and conditions of the license and applicable regulations. The Vice President – Operations is the Plant Manager. The Plant Manager has overall responsibility for safety and compliance to this policy. In particular, LES employs the principle of keeping radiation and chemical exposures to employees and the general public as low as reasonably achievable (ALARA).

The information provided in this chapter, the corresponding regulatory requirement, and the section of NUREG-1520, Chapter 2 in which the NRC acceptance criteria are presented is summarized below.

2.0 Organization and Administration

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 2 Reference
Section 2.1 Organizational Structure		
• Functional description of specific organization groups responsible for managing the design, construction, and operation of the facility	70.22(a)(6)	2.4.3
• Management controls and communications among organizational units	70.22(a)(8)	2.4.3
• Startup and transition to operations	70.22(a)(6)	2.4.3
Section 2.2 Key Management Positions		
• Qualifications, responsibilities, and authorities for key management personnel	70.22(a)(6)	2.4.3
Section 2.3 Administration		
• Effective implementation of HS&E functions using written procedures	70.22(a)(8)	2.4.3
• Reporting of unsafe conditions or activities	70.62(a)	2.4.3
• Commitment to establish formal management measures to ensure availability of IROFS	70.62(d)	2.4.3
• Written agreements with offsite emergency resources	70.22(i)	2.4.3

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.

Lea County, New Mexico has accepted the LES proposal to develop the NEF. Lea County has issued its Industrial Revenue Bond (National Enrichment Facility Project) Series 2004 in the maximum aggregate principal amount of $400,000,000 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 Vice President - Operations & Chief Nuclear Officer, the Vice President - Project, Vice President - Regulatory Affairs and General Counsel, and Chief Financial Officer. The Quality Assurance Manager reports to the Director of Compliance for functional day to day activities and has a direct line of communication to the Vice President – Operations & Chief Nuclear Officer and President for stop work authority. The Health Safety & Environmental Manager reports to the Director of Compliance. The HS&E Manager has a direct line of communication to the Vice President - Operations & Chief Nuclear Officer for all matters concerning safety during operations and. Figure 2.1-2, LES National Enrichment Facility Operating Organization shows the authority and lines of communication.

2.1.2 Project 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) was 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 was contracted to perform the site selection evaluation. A nuclear consulting company was 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 Project is responsible for managing, construction, construction turnover testing activities, and overall design responsibility. The Director of Engineering 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.

Urenco and ETC 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 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 which reports to the President.

The Vice President of 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 2.1-2.

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

2.1.3 Operating Organization

The operating organization for LES is shown in Figure 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 Chief Nuclear Officer and reports to the President. The Plant Manager is responsible for the overall operation and administration of the enrichment facility after formal turnover from Project 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
- Compliance

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 Project 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) has commenced operation with the first cascade (Cascade 101) being 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.

The focus of the organization has shifted from the project to construction turnover, initial start-up and operation of each facility system and subsequent cascades. LES has provided for staffing of the LES NEF Operating Organization to ensure smooth transition from construction activities to operation activities. The Health, Safety, and Environmental Manager and Director of Compliance have the authority to report safety concerns directly to the Vice President - Operations & Chief Nuclear Officer (as shown in Figure 2.1-2) for HS&E matters related to operations, design or construction. These positions are intentionally provided stop work authority at the Vice President - Operations & Chief Nuclear Officer level to provide significant continued focus on the health, safety, and environment goals during design, construction, and operations. URENCO, which has been operating gas centrifuge enrichment facilities in Europe for over 30 years, will have personnel integrated into the LES organization to provide technical support during startup of the facility and transition into the operations phase.

As the construction of systems is completed, the systems will undergo acceptance testing followed by turnover from the project organization to the operations organization. The turnover will include the physical systems and corresponding design information and records. Following turnover, the operating organization will be responsible for system maintenance and configuration management. The design basis for the facility is maintained during the transition from project to operations through the configuration management system described in Chapter 11, Management Measures.

2.2 Key Management Positions

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

The responsibilities, authorities and lines of communication for each key management position are provided in this section. Responsible managers have the authority to delegate tasks to other individuals; however, the responsible manager retains the ultimate responsibility and accountability for implementing the applicable requirements. Management responsibilities, supervisory responsibilities, and the criticality safety engineering staff responsibilities related to nuclear criticality safety are in accordance with ANSI/ANS-8.19, Administrative Practices for Nuclear Criticality Safety.

The LES Corporate and Operating Organization and lines of communication are shown in Figure 2.1-2.

2.2.1 Operating Organization

The functions and responsibilities of key facility management are described in the following paragraphs. Additional detailed responsibilities related to nuclear criticality safety for key management positions and remaining supervisory and criticality safety staff are in accordance with ANSI/ANS-8.19. Some position titles have been changed to better reflect the actual responsibilities of the position. Similarly, some operating functions have been assigned to different managers to better reflect the operating organization presently used at Urenco and U. S. nuclear facilities.

A. Vice President – Operations & Chief Nuclear Officer

The Vice President – Operations & Chief Nuclear Officer reports to the President and is a critical member of the leadership team for LES, with the ultimate responsibility for the nuclear safety, industrial safety, security, and operations of the facility. The Vice President – Operations & Chief Nuclear Officer is ultimately responsible for completion and safe operation of the NEF and has stop work authority for both the project and operations at the NEF.

The Vice President – Operations & Chief Nuclear Officer is also responsible for ensuring the facility complies with all applicable regulatory requirements. The Vice President – Operations & Chief Nuclear Officer is the Plant Manager. The Plant Manager has direct responsibility for operation of the facility in a safe, reliable and efficient manner. The Plant Manager is responsible for proper selection of staff for all key positions including positions on the Safety Review Committee. The Plant Manager is responsible for the protection of the facility staff and the general public from radiation and chemical exposure and/or any other consequences of an accident at the facility and also bears the responsibility for compliance with the facility license.

B. Deleted

C. Quality Assurance Manager

The Quality Assurance Manager reports to the Director of Compliance and has overall responsibility for the management and implementation of the LES QAPD.

2.2 Key Management Positions

The facility line managers and their staff who are responsible for performing quality-affecting work are responsible for ensuring implementation of and compliance with the QAPD. The QA Manager position maintains reporting relationship independence from management positions at the facility. The QA Manager has a direct relationship with the Vice President - Operations and Chief Nuclear Officer and President for quality concerns with Performance Assessment and Feedback or HS&E. The QA Manager has sufficient independence for all issues affecting quality. In addition the QA Manager has a reporting relationship with the Vice President - Operations and Chief Nuclear Officer and President for adequate stop work authority.

D. Health Safety and Environmental Manager

The Health Safety and Environmental Manager reports to the Director of Compliance and has the responsibility for assuring safety at the facility through activities including health and safety activities associated with nuclear criticality safety. The Health Safety and Environmental Manager works with the other facility managers to ensure consistent interpretations of health safety, and environmental requirements, performs independent reviews, and supports facility and operations change control reviews.

This position has a line of communications to the Vice President - Operations and Chief Nuclear Officer to ensure objective health, safety, and environmental audit, review, and control activities are maintained. This position is provided a reporting relationship to the Vice President – Operations & Chief Nuclear Officer level for stop work authority.

E. Operations Director

The Operations Director reports to the Plant Manager and has the responsibility for Shift Operations, Operations Support, Logistics Services, Chemistry and Environmental Services (Waste Analysis, Effluent Monitoring, and Product Assay), and Commissioning and Acceptance. This includes such activities as ensuring the correct and safe operation of UF_6 processes, proper handling of UF_6, and the identification and mitigation of any off normal operating conditions, UF_6 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. In the event of the absence of the Plant Manager, the Operations Director may assume the responsibilities and authorities of the Plant Manager.

F. Technical Services Director

The Technical Services Director reports to the Plant Manager and is the operations NEF Design Authority with responsibility for approving any modifications to operating portions of the facility (i.e., portions of the facility that have been formally turned over from the Project and accepted by Operations). The Technical Services Director assumes responsibility for all remaining NEF Design Authority responsibilities for the operating portions of the facility after formal turnover from the Project. NEF Design Authority responsibilities include approving design standards and design criteria, preparing and reviewing the NEF Functional Specification, leading the 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, Performance Assessment, Training, Administrative Support, and the Procedures Program. The Plant Support Director, in coordination with the Director, Human Resources & Communications, 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.

H. Commissioning & Acceptance Manager

The Commissioning & Acceptance Manager reports to the Operations Director and has the responsibility for the turnover and commissioning SSCs.

I. Performance Assessment and Feedback Manager

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

J. Quality Assurance Inspectors

The Quality Assurance Inspectors report to the Quality Assurance Manager (via a designated supervisory position, if applicable) and have the responsibility for performing inspections related to the implementation of the LES QAPD.

K. Quality Assurance Auditors

The Quality Assurance Auditors report to the Quality Assurance Manager (via a designated supervisory position, if applicable) and have the responsibility for performing audits related to the implementation of the LES QAPD.

L. Quality Assurance Technical Support

The Quality Assurance Technical Support personnel report to the Quality Assurance Manager (via a designated supervisory position, if applicable) and have the responsibility for providing technical support related to the implementation of the LES QAPD.

M. Emergency Preparedness Manager

The Emergency Preparedness Manager reports to the Plant Support Director and has the responsibility for ensuring the facility remains prepared to react and respond to any emergency situation that may arise. This includes emergency preparedness training of facility personnel, facility support personnel, the training of, and coordination with, offsite emergency response organizations (EROs), and conducting periodic drills to ensure facility personnel and offsite response organization personnel training is maintained up to date.

N. Director of Engineering

The Director of Engineering reports to the Vice President Project and has responsible for developing and managing the URENCO USA Project engineering personnel, procedures, processes and programs to ensure the construction and configuration of the facility is in accordance with the approved design and licensing basis. This position ensures Project engineering procedures and design products are adequately reviewed, approved and controlled to comply with license and regulatory requirements.

O. Environmental Compliance Officer

The Environmental Compliance Officer reports to the Health, Safety, and Environmental Manager and has the responsibility for coordinating facility activities to ensure all local, state and federal environmental regulations are met. This includes conducting the radiological environmental monitoring program and coordination of the submission of periodic reports with the Chemistry and Environmental Services organization to appropriate regulating organizations of effluents from the facility.

P. Radiation Protection Manager

The Radiation Protection Manager reports to the Director of Compliance and has the responsibility for implementing the Radiation Protection program. These duties include the training of personnel in use of equipment, control of radiation exposure of personnel, and continuous determination of the radiological status of the facility.

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 Director of Programs and Performance 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 Health, Safety and Environmental Manager 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, Safety, and Environmental Manager and is responsible for implementing the Criticality Safety Program in the operating organization, including ensuring that periodic nuclear criticality safety assessments are performed and reported.

T. Criticality Safety Engineers

Criticality Safety Engineers report to the Engineering Manager 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 UF_6 processes, proper handling of UF_6, and the identification and mitigation of any off normal operating conditions. In the event of the absence of the Plant Manager, the Shift Operations Manager may assume the responsibilities and authorities of the Plant Manager.

W. Shift Managers

The Shift Managers report to the Shift Operations Manager and have the responsibility for ensuring safe operation of enrichment equipment and support equipment. Each Shift Manager directs assigned personnel in order to provide enrichment services in a safe, efficient manner. In the event of the absence of the Plant Manager, the On-Duty Shift Manager may assume the responsibilities and authorities of the Plant Manager.

X. Safeguards Manager

The Safeguards Manager reports to the Director of Compliance and has the responsibility for ensuring the proper implementation of the FNMC Plan. This position is separate from and independent of the Operations, Technical Services, Construction and Performance Assessment and Feedback departments to ensure a definite division between the safeguards group and the other departments. In matters involving safeguards, the Director of Compliance, which the Safeguards Manager reports to, has direct access to the Vice President - Operations & Chief Nuclear Officer.

Y. Chemistry Services Manager

The Chemistry Services Manager reports to the Operations Director and has the responsibility for the implementation of chemistry analysis programs and procedures for the facility. Chemistry Analysis Activities includes effluent sample collection, chemical and radioactive analysis of effluents, comparison of effluent analysis results to limits, and reporting of chemical analysis of effluents to appropriate regulatory agencies.

Z. Logistics Services Manager

The Logistics Services Manager reports to the Director of Operations and is responsible for production planning, transport planning, uranium administration, safeguards operational support and materials handling, ensuring that cylinders of uranium hexafluoride are received and routed correctly at the facility, and all transportation licensing is properly implemented and maintained.

AA. Engineering Manager

The Engineering Manager reports to the Technical Services Director upon formal turnover of NEF Design Authority responsibilities from Project Engineering to the Technical Services Director. The Engineering Manager has the responsibility for providing engineering and technical support at the facility and maintaining the configuration management system. During the operations phase, the Engineering Manager is responsible for the development of all design changes to the plant and in support of the NEF Design Authority manages and controls the design basis. During all phases of design, construction and operation the Engineering Manager supports the NEF Design Authority by developing and maintaining the processes for design and configuration control and providing technical support for review of proposed changes to the approved design.

BB. Maintenance Manager

The Maintenance Manager reports to the Technical Services Director and has the responsibility of directing and scheduling maintenance activities to ensure proper operation of the facility, including preparation and implementation of maintenance, surveillance, and test procedures. This includes activities such as repair and preventive maintenance of facility equipment. The Maintenance Manager is responsible for plant systems availability and reliability as well as for 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 Technical Services Director and has the responsibility for adequately controlling documents at the facility.

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.

GG. Deleted

HH. Deleted

II. Director of Compliance

The Director of Compliance reports to the Plant Manager and has the responsibility for Quality Assurance, Operational Health, Safety, and Environmental responsibilities, Radiation Protection and Material Control and Accountability (Safeguards). This position ensures proper contamination control; has the responsibility for the submittal of NRC MC&A reports; and has overall responsibility for development of the LES QA Program.

This position reports to the Vice President – Operations and Chief Nuclear Officer (Plant Manager) to ensure objective nuclear safety audit, review, and control activities are maintained independent of the Operations Director. This position is intentionally provided a reporting relationship to the Vice President – Operations & Chief Nuclear Officer level for stop work authority.

JJ. Director of Programs and Performance

The Director of Programs and Performance leads the Programs and Performance department of the Project organization. This position encompasses the Corrective Action Program (CAP), Work Plans Management, Project Contracts and Industrial Safety for Construction, testing and turnover for various phases of the project. This position reports to the Vice President of Project.

2.2.2 Shift Crew Composition

The minimum operating shift crew consists of a Shift Manager (or Deputy Shift Manager in the absence of the Shift Manager), one Control Room operator, one operator for each SBM, security personnel, and one Radiation Protection Staff member or operator trained to monitor and perform routine radiological protection activities and certain, time-critical, radiation protective actions described in the NEF Emergency Plan. When only one SBM is in operation, a minimum of two operators is required.

At least one criticality safety engineer or the criticality safety officer will be available, with appropriate ability to be contacted by the Shift Manager, to respond to any routine request or emergency condition. This availability may be offsite if adequate communication ability is provided to allow response as needed.

2.2.3 Safety Review Committee

The facility maintains a Safety Review Committee (SRC) to assist with the safe operation of the facility. The SRC reports to the Plant Manager and provides technical and administrative review and audit of operations that could impact plant worker, public safety and environmental impacts. The scope of activities reviewed and audited by the SRC shall, as a minimum, include the following:

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

The SRC shall conduct at least one facility audit per year for the above areas.

The Safety Review Committee shall be composed of at least five members, including the Chairman. Members of the SRC may be from the LES corporate office or technical staff. The five members shall include experts on operations and all safety disciplines (criticality, radiological, chemical, industrial). The Chairman, members and alternate members of the Safety Review Committee shall be formally appointed by the Plant Manager, shall have an academic degree in an engineering or physical science field; and, in addition, shall have a minimum of five years of technical experience, of which a minimum of three years shall relate directly to one or more of the safety disciplines (criticality, radiological, chemical, industrial).

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 Vice President - Operations 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., 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 Vice President - Operations 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.

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

A. Deleted

B. Vice President – Operations & Chief Nuclear Officer

The President of LES, based on the individual's experience, proven ability in management of large scale facilities, and overall leadership qualities, appoints the Vice President - Operations & Chief Nuclear Officer.

This appointment by the President of LES reflects confidence in the individual's ability as effective programs, operations, regulatory, and business manager. The Vice President - Operations & Chief Nuclear Officer shall have, as a minimum, a bachelor's degree (or equivalent) and at least ten years related experience and/or training, or twenty years of related experience.

The Vice President – Operations & Chief Nuclear Officer is the Plant Manager, who is the overall manager of the facility. The Plant Manager shall be knowledgeable of the enrichment process, enrichment process controls and ancillary processes, criticality safety control, chemical safety, industrial safety, and radiation protection program concepts as they apply to the overall safety of a nuclear facility. The Plant Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and ten years of responsible nuclear experience.

C. Quality Assurance Manager

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

D. Health, Safety, and Environmental Manager

The Health, Safety, and Environmental Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and at least five years of responsible nuclear experience in HS&E or related disciplines. The Health, Safety, and Environmental Manager shall also have at least one year of experience/familiarity associated with nuclear criticality safety programs.

E. Operations Director

The Operations 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.

F. Shift Operations Manager

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

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. Director of Engineering

The Director of Engineering shall have a bachelor's degree in an engineering or science field and a minimum of 5 years of appropriate, responsible nuclear experience in implementing and supervising an engineering organization.

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.O.

O. Criticality Safety Engineers

The Criticality Safety Engineers 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, these individuals shall have at least two years of experience performing criticality safety analyses.

Should a change to the facility require a nuclear criticality safety evaluation or analysis, an individual who, as a minimum, possesses the equivalent qualifications of the Criticality Safety Engineer shall perform the evaluation or analysis. An independent review of the evaluation or analysis, shall be performed by a second Criticality Safety Engineer with the same minimum qualifications.

P. Deleted

Q. Shift Managers

Shift Managers shall have High School Diplomas (or equivalent) and a minimum of five years of appropriate operating experience at a nuclear or chemical process facility.

R. Logistics Services Manager

The Logistics Services Manager shall have, as a minimum, a bachelor's degree (or equivalent) and have a minimum of three years of appropriate, responsible experience in implementing and supervising a logistics program.

S. Safeguards Manager

The Safeguards Manager shall have as a minimum a bachelor's degree in an engineering or scientific field, and five years of experience in the management of a safeguards program for Special Nuclear Material, including responsibilities for material control and accounting. No credit for academic training may be taken toward fulfilling this experience requirement.

T. Chemistry Services Manager

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

U. Engineering Manager

The Engineering Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear engineering program.

V. Maintenance Manager

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

W. Security Manager

The Security Manager shall have a bachelor's degree (or equivalent) and five years of experience or an associates degree (or equivalent) and ten years of experience. Experience must be in the management of physical security at a facility requiring security capabilities similar to that required for the facility.

X. Training Manager

The Training Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a training program.

Y. Fire Protection Officer

The Fire Protection Officer shall have bachelor's degree (or equivalent) and shall be trained in the field of fire protection and have practical day-to-day experience at nuclear facilities.

Z. Information Services Manager

The Information Services Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a document control program.

AA. Performance Assessment and Feedback Manager

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

BB. Procurement Director

The Procurement Director shall have, as a minimum, a bachelor's degree (or equivalent) and have a minimum of three years of appropriate, responsible experience in implementing and supervising a procurement program.

CC. Deleted

DD. Deleted

EE. Director of Compliance

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

FF. Director of Programs and Performance

The Director of Programs and Performance shall have at a minimum, a Bachelor's degree (or equivalent) and have a minimum (12) years related management or leadership experience associated with Programs and Performance indicators at a Nuclear or Chemical Plant Processing facility or related start-up construction experience of which 4 years may be substituted for a Bachelors degree.

2.3 Administration

This section summarizes how the activities that are essential for implementation of the management measures and other HS&E functions are documented in formally approved, written procedures, prepared in compliance with a formal document control program. The mechanism for reporting potentially unsafe conditions or activities to the Plant Support organization and facility management is also summarized. Details of the management measures are provided in Chapter 11, Management Measures.

2.3.1 Configuration Management

Configuration management is provided for Items Relied On For Safety (IROFS) throughout facility design, construction, testing, and operation. Configuration management provides the means to establish and maintain a technical baseline for the facility based on clearly defined requirements. During design, construction, and operations (until formal turnover to the Technical Services Director), –Project Engineering has responsibility for configuration management through the design control process. Selected documentation is controlled under the configuration management system in accordance with appropriate QA procedures associated with design control, document control, and records management. Design changes to IROFS undergo formal review, including interdisciplinary reviews as appropriate, in accordance with these procedures.

Configuration management provides the means to establish and maintain the essential features of the design basis of IROFS. As the project progresses from design and construction to operation, configuration management is maintained by the facility engineering organization as the overall focus of activities changes.

Additional details on Configuration Management are provided in Chapter 11, Management Measures.

2.3.2 Maintenance

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

The purpose of planned and scheduled maintenance for IROFS is to ensure that the equipment and controls are kept in a condition of readiness to perform the planned and designed functions when required. Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance function is administratively closely coupled to operations. The maintenance organization plans, schedules, tracks, and maintains records for maintenance activities.

Maintenance activities generally fall into the following categories:

- Corrective maintenance
- Preventive maintenance
- Surveillance/monitoring

- 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 a 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.

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.

Policies and procedures will be developed to ensure that there are ties between major plant safety functions such as the ISA, management measures for items relied on for safety (IROFS), radiation safety, nuclear criticality safety, fire safety, chemical safety, environmental monitoring, and emergency planning.

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

2.3.5 Audits and Assessments

The LES QA Program requires periodic audits to confirm that activities affecting quality comply with the QAPD and that the QAPD is being implemented effectively. Also included in the QAPD are requirements to perform periodic Management Assessments.

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

2.3.5.1 Safety Review Committee

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

2.3.5.2 Quality Assurance Department

The Quality Assurance Department conducts periodic audits of activities associated with the facility, in order to verify the facility's compliance with established procedures in accordance with the QAPD. The LES Quality Assurance Program Description is included in Chapter 11, Management Measures as Appendix A.

2.3.5.3 Facility Operating Organization

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

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

2.3.5.4 Audited Organizations

Audited organizations shall assure that findings are evaluated and corrected in a timely manner in accordance with the QAPD Sections 16, Corrective Action and 18, Audits.

2.3.6 Incident Investigations

The Corrective Action Program (CAP) is described in detail in Section 11.6 and the QAPD Section 16, Corrective Action. Each event is considered in terms of its requirements for reporting in accordance with regulations and is evaluated to determine the level of investigation required. These evaluations and investigations are conducted in accordance with approved CAP procedures. The depth of the investigation depends upon the severity of the incident in terms of the levels of uranium released and/or the degree of potential for exposure of workers, the public or the environment.

2.3.7 Employee Concerns

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

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

- line management or other facility management (e.g., Performance Assessment and Feedback Management, Plant Manager, HS&E Manager, Plant Support Director, Director of Compliance
- the facility safety organization (i.e., any of the safety engineers or managers)
- NRC's requirements under 10 CFR 19, Notices, Instructions and Reports to Workers: Inspection and Investigations (CFR, 2003a)
- LES CAP - a simple mechanism available for use by any person at the NEF site for reporting unusual events and potentially unsafe conditions or activities.

2.3.8 Records Management

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

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

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

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

Additional details on the records management program are provided in Chapter 11, Management Measures.

2.3.9 Written Agreements with Offsite Emergency Resources

The plans for coping with emergencies at the facility are presented in detail in the Emergency Plan. The Emergency Plan includes a description of the facility emergency response organization and interfaces with off-site EROs. Written agreements between the facility and off-site EROs, including the local fire department, the local law enforcement agency, ambulance/rescue units, and medical services and facilities have been established.

Coordination with participating government agencies (State, Counties) is vital to the safety and health of plant personnel and the general public. The principal state and local agencies/organizations having responsibilities for radiological or other hazardous material emergencies for the facility are:

A. New Mexico Department of Public Safety
B. New Mexico Department of Homeland Security and Emergency Management
C. Eunice Emergency Response Services
D. Hobbs Emergency Response Services

Details of the interfaces with these agencies are provided in Section 4 of the Emergency Plan.

2.4 References

Edition of Codes, Standards, NRC Documents, etc that are not listed below are given in ISAS Table 3.0-1.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 19, Notices, Instructions and Reports to Workers: Inspection and Investigations, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Part 40, Domestic Licensing of Source Material, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

NRC, 1992. Proposed Method for Regulating Major Materials Licensees, NUREG-1324, U.S. Nuclear Regulatory Commission, 1992.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

2.5 Chapter 2 Figures

2.5 Chapter 2 Figures

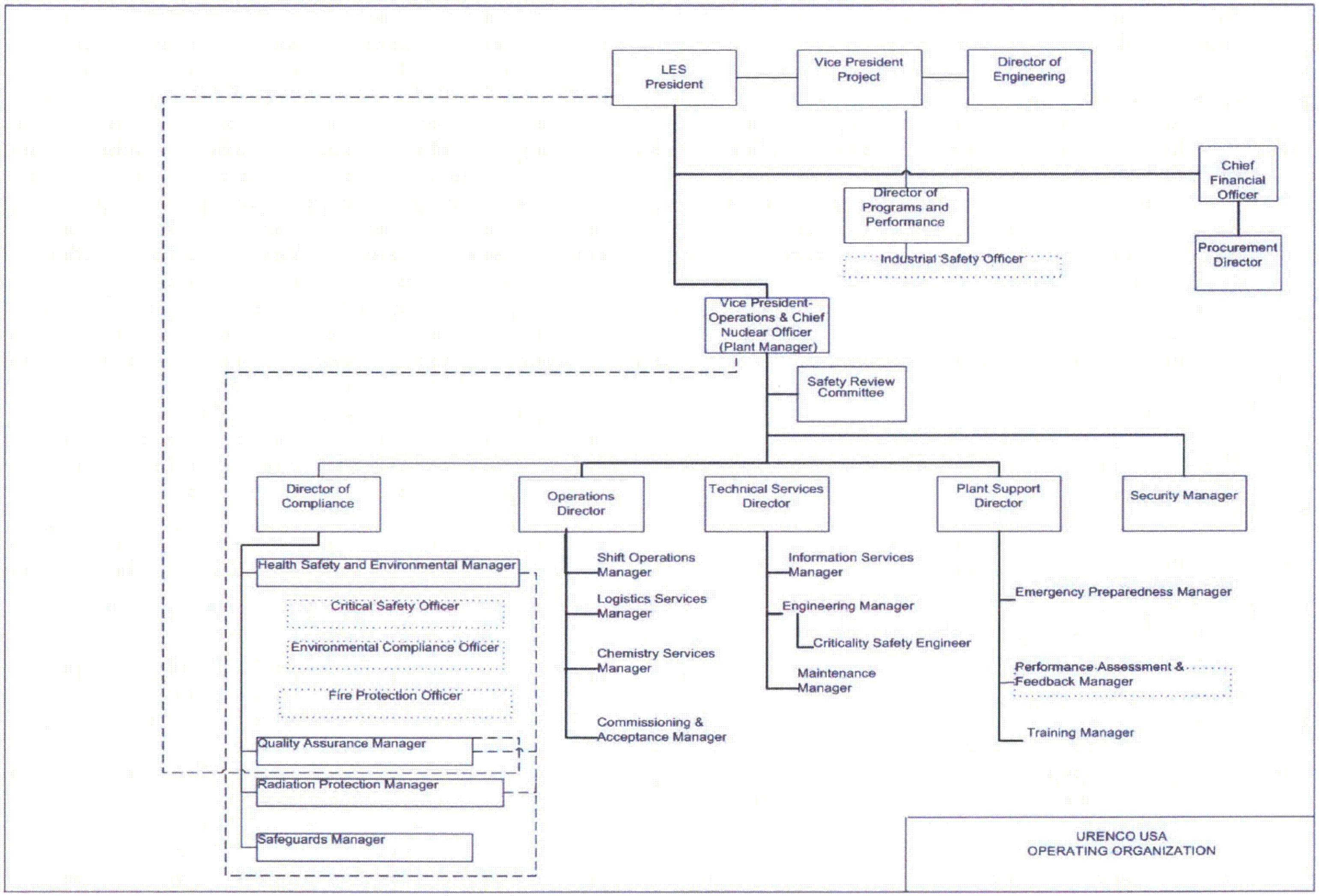


Figure 2.1-2 LES National Enrichment Facility Operating Organization

2.5 Chapter 2 Figures

3.0 Safety Program Commitments

This section presents the commitments pertaining to the facility's safety program including the performance of an ISA. 10 CFR Part 70 (CFR, 2003b) contains a number of specific safety program requirements related to the integrated safety analysis (ISA). These include the primary requirements that an ISA be conducted, and that it evaluate and show that the facility complies with the performance requirements of 10 CFR 70.61 (CFR, 2003c).

3.1 Safety Program

The three elements of the safety program defined in 10 CFR 70.62(a) (CFR, 2003d) are addressed below.

3.1.1 Process Safety Information

A. LES has compiled and maintains up-to-date documentation of process safety information. Written process-safety information is used in updating the ISA and in identifying and understanding the hazards associated with the processes. The compilation of written process-safety information includes information pertaining to:

1. The hazards of all materials used or produced in the process, which includes information on chemical and physical properties such as are included on Material Safety Data Sheets meeting the requirements of 29 CFR 1910.1200(g) (CFR, 2003e).
2. Technology of the process which includes block flow diagrams or simplified process flow diagrams, a brief outline of the process chemistry, safe upper and lower limits for controlled parameters (e.g., temperature, pressure, flow, and concentration), and evaluation of the health and safety consequences of process deviations.
3. Equipment used in the process including general information on topics such as the materials of construction, piping and instrumentation diagrams (P&IDs), ventilation, design codes and standards employed, material and energy balances, engineered IROFS, equipment essential to support administrative IROFS, electrical classification, and relief system design and design basis.

The process-safety information described above is maintained up-to-date by the configuration management program described in Section 11.1, Configuration Management.

B. LES has developed procedures and criteria for changing the ISA. This includes implementation of a facility change mechanism that meets the requirements of 10 CFR 70.72 (CFR, 2003f).

The development and implementation of procedures is described in Section 11.4, Procedures Development and Implementation.

C. LES uses personnel with the appropriate experience and expertise in engineering and process operations to maintain the ISA. The ISA Team for the various processes consists of individuals who are knowledgeable in the ISA method(s) and the operation, hazards, and safety design criteria of the particular process. Training and qualifications of individuals responsible for maintaining the ISA are described in Section 11.3, Training and Qualifications, Section 2.2, Key Management Positions, and Section 3.2, Integrated Safety Analysis Team.

3.1.2 Integrated Safety Analysis

A. LES has conducted an ISA for each process, such that it identifies (i) radiological hazards, (ii) chemical hazards that could increase radiological risk, (iii) facility hazards that could increase radiological risk, (iv) potential accident sequences, (v) consequences and likelihood of each accident sequence and (vi) IROFS including the assumptions and conditions under which they support compliance with the performance requirements of 10 CFR 70.61 (CFR, 2003c).

A synopsis of the results of the ISA, including the information specified in 10 CFR 70.65(b) (CFR, 2003a), is provided in the National Enrichment Facility Integrated Safety Analysis Summary.

B. LES has implemented programs to maintain the ISA and supporting documentation so that it is accurate and up-to-date. Changes to the ISA Summary are submitted to the NRC, in accordance with 10 CFR 70.72(d)(1) and (3) (CFR, 2003f). The ISA update process accounts for any changes made to the facility or its processes. This update will also verify that initiating event frequencies and IROFS reliability values assumed in the ISA remain valid. Any changes required to the ISA as a result of the update process will be included in a revision to the ISA. Management policies, organizational responsibilities, revision time frame, and procedures to perform and approve revisions to the ISA are outlined in Chapter 11.0, Management Measures. Evaluation of any facility changes or changes in the process safety information that may alter the parameters of an accident sequence is by the ISA method(s) as described in the ISA Summary Document. For any revisions to the ISA, personnel having qualifications similar to those of ISA team members who conducted the original ISA are used.

C. Personnel used to update and maintain the ISA and ISA Summary are trained in the ISA method(s) and are suitably qualified. Training and Qualification of personnel used to update or maintain the ISA are described in Section 11.3, Training and Qualifications.

D. Proposed changes to the facility or its operations are evaluated using the ISA method(s). New or additional IROFS and appropriate management measures are designated as required. The adequacy of existing IROFS and associated management measures are promptly evaluated to determine if they are impacted by changes to the facility and/or its processes. If a proposed change results in a new type of accident sequence or increases the consequences or likelihood of a previously analyzed accident sequence within the context of 10 CFR 70.61 (CFR, 2003c), the adequacy of existing IROFS and associated management measures are promptly evaluated and the necessary changes are made, if required.

E. Unacceptable performance deficiencies associated with IROFS are addressed that are identified through updates to the ISA.

F. Written procedures are maintained on site. Section 11.4, Procedures Development and Implementation, discusses the procedures program.

G. All IROFS are maintained so that they are available and reliable when needed.

3.1.3 Management Measures

Management measures are functions applied to IROFS, and any items that are essential to the function of IROFS. IROFS management measures ensure compliance with the performance requirements assumed in the ISA documentation. The measures are applied to particular structures, systems, equipment, components, and activities of personnel, and may be graded commensurate with the reduction of the risk attributable to that IROFS. The IROFS management measures shall ensure that these structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary are designed, implemented, and maintained, as necessary, to ensure they are available and reliable to perform their function when needed, to comply with the performance requirements assumed in the ISA documentation.

The following types of management measures are required by the 10 CFR 70.4 (CFR, 2003b) definition of management measures. The description for each management measure reflects the general requirements applicable to each IROFS. Any management measure that deviates from the general requirements described in this section, which are consistent with the performance requirements assumed in the ISA documentation, are discussed in the National Enrichment Facility Integrated Safety Analysis Summary.

Configuration Management

The configuration management program is required by 10 CFR 70.72 (CFR, 2003f) and establishes 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 are essential to the function of IROFS, is applied to all items identified within the scope of the IROFS boundary. Any change to structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary must be evaluated before the change is implemented. If the change requires an amendment to the License, Nuclear Regulatory Commission approval is required prior to implementation.

Maintenance

Maintenance of engineered IROFS, and any items that are essential to the function of IROFS, encompasses planned surveillance testing and preventative maintenance, as well as unplanned corrective maintenance. Implementation of approved configuration management changes to hardware is also generally performed as a planned maintenance function.

Planned surveillance testing (e.g., functional/performance testing, instrument calibrations) monitors the integrity and capability of IROFS, and any items that are essential to the function of IROFS, to ensure they are available and reliable to perform their function when needed, to comply with the performance requirements assumed in the ISA documentation. All necessary periodic surveillance testing is generally performed on an annual frequency (any exceptions credited within the ISA are discussed in the National Enrichment Facility Integrated Safety Analysis Summary).

Planned preventative maintenance (PM) includes periodic refurbishment, partial or complete overhaul, or replacement of engineered IROFS, as necessary, to ensure the continued availability and reliability of the safety function assumed in the ISA documentation. In determining the frequency of any PM, consideration is given to appropriately balancing the objective of preventing failures through maintenance, against the objective of minimizing unavailability of IROFS because of PM. In addition, feedback from PM and corrective maintenance and the results of incident investigations and identified root causes are used, as appropriate, to modify the frequency or scope of PM.

Planned maintenance on engineered IROFS, or any items that are essential to 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.

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 are essential to 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 are essential to 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).

Procedures

All activities involving IROFS, and any items that are essential to the function of IROFS, are conducted in accordance with approved procedures. Each of the other IROFS management measures (e.g., configuration management, maintenance, training) is implemented via approved procedures. These procedures are intended to provide a pre-planned method of conducting the activity in order to eliminate errors due to on-the-spot analysis and judgments.

All procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. However, written procedures cannot address all contingencies and operating conditions. Therefore, they contain a degree of flexibility appropriate to the activities being performed. Procedural guidance exists to identify the manner in which procedures are to be implemented. For example, routine procedural actions may not require the procedure to be present during implementation of the actions, while complex jobs, or checking with numerous sequences may require valve alignment checks, approved operator aids, or in-hand procedures that are referenced directly when the job is conducted.

To support the requirement to minimize challenges to IROFS, and any items that are essential to the function of IROFS, specific procedures for abnormal events are also provided. These procedures are based on a sequence of observations and actions to prevent or mitigate the consequences of an abnormal situation.

Audits and Assessments

Audits are focused on verifying compliance with regulatory and procedural requirements and licensing commitments. Assessments are focused on effectiveness of activities and ensuring that IROFS are reliable and are available to perform their intended safety functions as documented in the ISA. The frequency of audits and assessments is based upon the status and safety importance of the activities being performed and upon work history. However, at a minimum, all activities associated with maintaining IROFS will generally be audited or assessed on an annual basis (any exceptions credited within the ISA are discussed in the National Enrichment Facility Integrated Safety Analysis Summary).

Incident Investigations

Incident investigations are conducted within the Corrective Action Program (CAP). Incidents associated with IROFS, and any items that are essential to the function of IROFS, encompass a range of items, including (a) processes that behave in unexpected ways, (b) procedural activities not performed in accordance with the approved procedure, (c) discovered deficiency, degradation, or non-conformance with an IROFS, or any items that are essential to the function of IROFS. Additionally, audit and assessment results are tracked in the Corrective Action Program.

Feedback from the results of incident investigations and identified root causes are used, as appropriate, to modify management measures to provided continued assurance that the reliability and availability of IROFS remain consistent with the performance requirements assumed in the ISA documentation.

Records Management

All records associated with IROFS, and any items that are essential to the function of IROFS, shall be managed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures are included.

Other Quality Assurance Elements

Other quality assurance elements associated with IROFS, or any items that are essential to the function of IROFS, that are required to ensure the IROFS is available and reliable to perform the function when needed to comply with the performance requirements assumed in the ISA documentation, 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 (UF_6) 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/even 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 [e.g., What-If/Checklist. Failure Modes and Effects Analysis (FMEA), Fault Tree Analysis, Event Tree Analysis, Cause-Consequence Analysis, Human Reliability Analysis, etc.] that may be implemented. The guidance from NUREG-1513 will be followed for selection of a hazard analysis method. Furthermore, any hazard analysis method may be used as described in, and in accordance with, NUREG-1513. Thus, the hazard analysis methods that may be used are not limited to those briefly described in this ISA Methods section.

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.

3.2 Integrated Safety Analysis Methods

The tabular accident summary resulting from the ISA identifies, for each sequence, which engineered or administrative IROFS must fail to allow the occurrence of consequences that exceed the levels identified in 10 CFR 70.61 (CFR, 2003c).

For this license application, two ISA Teams were formed. This was necessary because the sensitive nature of some of the facility design information related to the enrichment process required the use of personnel with the appropriate national security clearances. This team performed the ISA on the Cascade System, Contingency Dump System, Centrifuge Test System and the Centrifuge Post Mortem System. This ISA Team is referred to as the Classified ISA Team. The Non-Classified Team, referred to in the remainder of this text as the ISA Team, performed the ISA on the remainder of the facility systems and structures. In addition, the (non-classified) ISA Team performed the External Events and Fire Hazard Assessment for the entire facility.

In preparing for the ISA, the Accident Analysis in the Safety Analysis Report (LES, 1993) for the Claiborne Enrichment Center was reviewed. In addition, experienced personnel with familiarity with the gas centrifuge enrichment technology safety analysis where used on the ISA Team. This provides a good peer check of the final ISA results.

A procedure was developed to guide the conduct of the ISA. This procedure was used by both teams. In addition, there were common participants on both teams to further integrate the approaches employed by both teams. These steps were taken to ensure the consistency of the results of the two teams. A non-classified summary of the results of the Classified ISA has been prepared and incorporated into the ISA Summary.

3.2.1 Hazard Identification

The hazard and operability (HAZOP) analysis method was used for identifying the hazards for the Uranium Hexafluoride (UF_6) process systems and Cylinder Receipt and Dispatch Building systems. This method is consistent with the guidance provided in NUREG-1513 and NUREG-1520. The hazards identification process results in identification of physical, radiological or chemical characteristics that have the potential for causing harm to site workers, the public, or to the environment. Hazards are identified through a systematic review process that entails the use of system descriptions, piping and instrumentation diagrams, process flow diagrams, plot plans, topographic maps, utility system drawings, and specifications of major process equipment. In addition, criticality hazards identification were performed for the areas of the facility where fissile material is expected to be present. The criticality safety analyses contain information about the location and geometry of the fissile material and other materials in the process, for both normal and credible abnormal conditions. The ISA input information is included in the ISA documentation and is available to be verified as part of an on-site review.

The hazard identification process documents materials that are:

- Radioactive
- Fissile
- Flammable
- Explosive
- Toxic

- Reactive.

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

For the purpose of evaluating the impacts of fire hazards, the ISA team considered the following:

- Postulated the development of a fire occurring in in-situ combustibles from an unidentified ignition source (e.g., electrical shorting, or other source)
- Postulated the development of a fire occurring in transient combustibles from an unidentified ignition source (e.g., electrical shorting, or other source)
- Evaluated the uranic content in the space and its configuration (e.g., UF_6 solid/gas in cylinders, UF_6 gas in piping, UF_6 and/or byproducts bound on chemical traps, Uranyl Fluoride (UO_2F_2) particulate on solid waste or in solution). The appropriate configuration was considered relative to the likelihood of the target releasing its uranic content as a result of a fire in the area.

In order to assess the potential severity of a given fire and the resulting failures to critical systems, the facility Fire Hazard Analysis was consulted. However, since the design supporting the license submittal for this facility is not yet at the detailed design stage, detailed in-situ combustible loading and in-situ combustible configuration information is not yet available. Therefore, in order to place reasonable and conservative bounds on the fire scenarios analyzed, the ISA Team estimated in-situ combustible loadings based on information of the in-situ combustible loading from Urenco's Almelo SP-5 plant (on which the National Enrichment Facility (NEF) design is based). This information from SP-5 indicates that in-situ combustible loads are expected to be very low.

The Fire Safety Management Program will limit the allowable quantity of transient combustibles in critical plant areas (i.e., uranium areas). Nevertheless, the ISA Team still assumed the presence of moderate quantities of ordinary (Class A) combustibles (e.g., trash, packing materials, maintenance items or packaging, etc.) in excess of anticipated procedural limits. This was not considered a failure of the associated administrative IROFS feature for controlling/ minimizing transient combustible loading in all radiation/uranium areas. Failure of the IROFS is connoted as the presence of extreme or severe quantities of transients (e.g., large piles of combustible solids, bulk quantities of flammable/combustible liquids or gases, etc.). The Urenco ISA Team representatives all indicated that these types of transient combustible conditions do not occur in the European plants. Accordingly, and given the orientation and training that facility employees will receive indicating that these types of fire hazards are unacceptable, the administrative IROFS preventing severe accumulations has been assigned a high degree of reliability.

Fires that involve additional in-situ or transient combustibles from outside each respective fire area could result in exposure of additional uranic content being released in a fire beyond the quantities assumed above. For this reason, fire barriers are needed to ensure that fires cannot propagate from non-uranium containing areas into uranium (U) areas or from one U area to another U area (unless the uranium content in the space is insignificant, i.e., would be a low consequence event). Fire barriers shall be designed with adequate safety margin such that the total combustible loading (in-situ and transient) allowed to expose the barrier will not exceed 80% of the hourly fire resistance rating of the barrier.

For external events, the impacts were evaluated for the following hazards:

External events were considered at the site and facility level versus at individual system nodes. Specific external event HAZOP guidewords were developed for use during the external event portion of the ISA. The external event ISA considered both natural phenomena and man-made hazards. During the external event ISA team meeting, each area of the plant was discussed as to whether or not it could be adversely affected by the specific external event under consideration. If so, specific consequences were then discussed. If the consequences were known or assumed to be high, then a specific design basis with a likelihood of highly unlikely would be selected.

Given that external events were considered at the facility level, the ISA for external events was performed after the ISA team meetings for all plant systems were completed. This provided the best opportunity to perform the ISA at the site or facility level. Each external event was assessed for both the uncontrolled case and then for the controlled case. The controlled cases could be a specific design basis for that external event, IROFS or a combination of both. An Accident Sequence and Risk matrix was prepared for each external event.

External events evaluated included:

- Seismic
- Tornado, Tornado Missile and High Wind
- Snow and Ice
- Flooding
- Local Precipitation
- Other (Transportation and Nearby Facility Accidents)
- Aircraft
- Pipelines
- Highway
- Other Nearby Facilities
- Railroad
- Internal Flooding from On-Site Above Ground Liquid Storage Tanks.

The ISA is intended to give assurance that the potential failures, hazards, accident sequences, scenarios, and IROFS have been investigated in an integrated fashion, so as to adequately consider common mode and common cause situations. Included in this integrated review is the identification of IROFS function that may be simultaneously beneficial and harmful with respect to different hazards, and interactions that might not have been considered in the previously completed sub-analyses. This review is intended to ensure that the designation of one IROFS does not negate the preventive or mitigation function of another IROFS. An integration checklist is used by the ISA Team as a guide to facilitate the integrated review process.

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 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 UF_6 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:

- The ISA Team created a list of deviations for the UF_6 process, other processes in which the deviation could potentially impact the UF_6 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.

- In addition to identification of safeguards, the ISA Team also considered any existing design features that could mitigate/reduce the consequences.
- 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).

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.
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
- 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

3.2.5 Risk Matrix Development

3.2.5.1 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 UO_2F_2, 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.

3.2.5.2 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.

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_{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 thickness. A set of generic conservative criticality calculations has determined the maximum volume, diameter, or slab thickness (i.e., safe value) that would result in a k_{eff} < 0.95. A component in this category has a volume, diameter or slab thickness that is less than the associated safe value resulting from the generic conservative criticality calculations and therefore the k_{eff} associated with this component is < 0.95. The components in the second category require a more detailed criticality analysis (i.e., a criticality analysis of the physical arrangement of the component's design configuration) to show that k_{eff} is < 0.95. In the second category of components, the design configuration is not bounded by the results of the generic conservative criticality calculations for maximum volume, diameter, or slab thickness that would result in a k_{eff} < 0.95. Examples of components in this second category are the product pumps that have volumes greater than the safe-by-volume value, but are shown by specific criticality analysis to have a k_{eff} < 0.95.

For failure of passive safe-by-design components to be considered "highly unlikely," these components must also meet the criterion that the only potential means to effect a change that might result in a failure to function, would be to implement a design change (i.e., geometry deformation as a result of a credible process deviation or event does not adversely impact the performance of the safety function). The evaluation of the potential to adversely impact the safety function of these passive design features includes consideration of potential mechanisms to cause bulging, corrosion, and breach of confinement/leakage and subsequent accumulation of material. The evaluation further includes consideration of adequate controls to ensure that the double contingency principle is met. For each of these passive design components, it must be concluded, that there is no credible means to effect a geometry change that might result in a failure of the safety function and that significant margin exists. For components that are safe-by-volume, safe-by-diameter, or safe-by-slab thickness (i.e., first category of safe-by-design components), significant margin is defined as a margin of at least 10%, during both normal and upset conditions, between the actual design parameter value of the component and the value of the corresponding critical design attribute. For components that require a more detailed criticality analysis (i.e., second category of safe-by-design components), significant margin is defined as $k_{eff} < 0.95$, where $k_{eff} = kcalc + 3\sigma calc$. This margin is considered acceptable since the calculation of k_{eff} also conservatively assumes the components are full of uranic breakdown material at maximum enrichment, the worst credible moderation conditions exist, and the worst credible reflection conditions exist. In addition, the configuration management system required by 10 CFR 70.72 (implemented by the NEF Configuration Management Program) ensures the maintenance of the safety function of these features and assures compliance with the double contingency principle, as well as the defense-in-depth criterion of 10 CFR 70.64(b).

Guidance from revisions of issued versions of NUREG 1520 was used in creating the definition of "not credible." If an event is not credible, IROFS are not required to prevent or mitigate the event. The fact that an event is not "credible" must not depend on any facility feature that could credibly fail to function. One cannot claim that a process does not need IROFS because it is "not credible" due to characteristics provided by IROFS. The implication of "credible" in 10 CFR 70.61 (CFR, 2003c) is that events that are not "credible" may be neglected.

Any one of the following independent acceptable sets of qualities could define an event as not credible:

a. An external event for which the frequency of occurrence can conservatively be estimated as less than once in a million years
b. A process deviation that consists of a sequence of many unlikely events or errors for which there is no reason or motive. In determining that there is no reason for such errors, a wide range of possible motives, short of intent to cause harm, must be considered. Complete ignorance of safe procedures is possible for untrained personnel, which should be considered a credible possibility. Obviously, no sequence of events should be categorized as not credible if it has actually occurred 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.

3.2.5.3 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.

3.2.6 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 assigned to one of the three likelihood categories of the risk matrix, depending on the value of this index in accordance with Table 3.1-8, Determination of Likelihood Category.

The values of index numbers in accident sequences are assigned considering the criteria in Tables 3.1-9 through 3.1-11. Each table applies to a different type of event. Table 3.1-9, Failure Frequency Index Numbers, applies to events that have frequencies of occurrence, such as initiating events and certain IROFS failures. Failure Probability Index Numbers are evaluated based on operating experience, (either from Urenco or the National Enrichment Facility, as appropriate) or analyses. When failure probabilities are required for an event, Table 3.1-10, Failure Probability Index Numbers, provides the index values. Table 3.1-11, Failure Duration Index Numbers, provides index numbers for durations of failure. These are used in certain accident sequences where two IROFS must simultaneously be in a failed state. In this case, one of the two controlled parameters will fail first. It is then necessary to consider the duration that the system remains vulnerable to failure of the second. This period of vulnerability can be terminated in several ways. The first failure may be "fail-safe" or be continuously monitored, thus alerting the operator when it fails so that the system may be quickly placed in a safe state. Or the IROFS may be subject to periodic surveillance tests for hidden failures. When hidden failures are possible, these surveillance intervals limit the duration that the system is in a vulnerable state. The reverse sequences, where the second IROFS fails first, should be considered as a separate accident sequence. This is necessary because the failure frequency and the duration of outage of the first and the second IROFS may differ. The values of these duration indices are not merely judgmental. They are directly related to the time intervals used for surveillance and the time needed to render the system safe.

The duration of failure is accounted for in establishing the overall likelihood that an accident sequence will continue to the defined consequence. Thus, the time to discover and repair the failure is accounted for in establishing the risk of the postulated accident.

The total likelihood index is the sum of the indices for all the events in the sequence, including those for duration. Consequences are assigned to one of the three consequence categories of the risk matrix, based on calculations or estimates of the actual consequences of the accident sequence.. The consequence categories are based on the levels identified in 10 CFR 70.61 (CFR, 2003c). Multiple types of consequences can result from the same event. The consequence category is chosen for the most severe consequence.

In summarizing the ISA results, Table 3.7-1, Accident Sequence and Risk Index, provides two risk indices for each accident sequence to permit evaluation of the risk significance of the IROFS involved. To measure whether an IROFS has high risk significance, the table provides an "uncontrolled risk index," determined by modeling the sequence with all IROFS as failed

(i.e., not contributing to a lower likelihood). In addition, a "controlled risk index" is also calculated, taking credit for the low likelihood and duration of IROFS failures. When an accident sequence has an uncontrolled risk index exceeding four but a controlled risk index of less than four, the IROFS involved have a high risk significance because they are relied on to achieve acceptable safety performance. Thus, use of these indices permits evaluation of the possible benefit of improving IROFS and also whether a relaxation may be acceptable.

3.3 Integrated Safety Analysis Team

There were two ISA Teams that were employed in the ISA. The first team worked on the non-classified portions of the facility and is referred to in the text as the ISA Team. The second team, referred to as the Classified ISA Team, performed the ISA on the classified elements of the facility. Both teams were selected with credentials consistent with the requirements in

10 CFR 70.65 (CFR, 2003a) and the guidance provided in NUREG-1520. To facilitate consistency of results, common membership was dictated as demonstrated below

(i.e., some members of the Non-Classified Team participated on the Classified Team. One of the members of the Classified Team participated in the ISA Team Leader Training, which was conducted prior to initiating the ISA. In addition, the Classified ISA Team Leader observed some of the non-classified ISA Team meetings.

The ISA was performed by a team with expertise in engineering, safety analysis and enrichment process operations. The team included personnel with experience and knowledge specific to each process or system being evaluated. The team was comprised of individuals who have experience, individually or collectively, in:

- Nuclear criticality safety
- Radiological safety
- Fire safety
- Chemical process safety
- Operations and maintenance
- ISA methods.

The ISA team leader was trained and knowledgeable in the ISA method(s) chosen for the hazard and accidents evaluations. Collectively, the team had an understanding of all process operations and hazards under evaluation.

The ISA Manager was responsible for the overall direction of the ISA. The process expertise was provided by the Urenco personnel on the team. In addition, the Team Leader has an adequate understanding of the process operations and hazards evaluated in the ISA, but is not the responsible cognizant engineer or enrichment process expert.

3.4 Compliance Item Commitments

3.4.1 For accident sequences PT3-5, FR1-1, FR1-2, FR2-1, FR2-2, DS1-1, DS1-2, DS2-1, DS2-2, DS3-1, DS3-2, SW1-1, SW1-2, LW1-2, LW1-3, RD1-1, and EC3-1, an Initiating Event Frequency (IEF) index number of "-2" may be assigned based on evidence from the operating history of similar designed Urenco European plants. Detailed justifications for the IEF index numbers of "-2" will be developed during detailed design. If the detailed justification does not support the IEF index number of "-2," then the IEF index number assigned and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with overall ISA methodology.

3.4.2 For Administrative Control IROFS that involve "use of" a component or device, a Failure Probability Index Number (FPIN) of "-2" may be assigned provided the IROFS is a routine, simple, action that either: (1) involves only one or two decision points or (2) is highly detailed in the associated implementing procedure. Alternately, an FPIN of "-3" may be assigned for this type of IROFS provided the criteria specified above for an FPIN of "-2" are met and the IROFS is enhanced by requiring independent verification of the safety function. This enhancement shall meet the requirements for independent verification identified in item 3.4.5 below. If these criteria cannot be met, then the FPIN assigned to the IROFS and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with the overall ISA methodology.

3.4.3 For Administrative Control IROFS that involve "verification of" a state or condition, an FPIN of "-2" may be assigned provided the IROFS is a routine action performed by one person, with proceduralized, objective, acceptance criteria. Alternately, an FPIN of "-3" may be assigned for this type of IROFS provided the criteria specified above for an FPIN of "-2" are met and the IROFS is enhanced by requiring independent verification of the safety function. This enhancement shall meet the requirements for independent verification identified in item 3.4.5 below. If these criteria cannot be met, then the FPIN assigned to the IROFS and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with the overall ISA methodology.

3.4.4 For Administrative Control IROFS that involve "independent sampling," different samples are obtained and an FPIN of "-2" may be assigned provided at least three of the following four criteria are met.

1. Different methods/techniques are used for sample analysis.
2. Samples are obtained from different locations.
3. Samples are obtained at different times. The time period between collection of the different samples shall be sufficient to ensure results are meaningful and representative of the material sampled.
4. Samples are obtained by different personnel.

If at least three of the above criteria cannot be met, then the FPIN assigned to the IROFS and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with the overall ISA methodology.

3.4.5 **For IROFS and IROFS with Enhanced Failure Probability Index Numbers** (i.e., enhanced IROFS) that require "independent verification" of a safety function, the independent verification shall be independent with respect to personnel and personnel interface. Specifically, a second qualified individual, operating independently (e.g., not at the same time or not at the same location) of the individual assigned the responsibility to perform the required task, shall, as applicable, verify that the required task (i.e., safety function) has been performed correctly (e.g., verify a condition), or re-perform the task (i.e., safety function), and confirm acceptable results before additional action(s) can be taken which potentially negatively impact the safety function of the IROFS. The required task and independent verification shall be implemented by procedure and documented by initials or signatures of the individuals responsible for each task. In addition, the individuals performing the tasks shall be qualified to perform, for the particular system or process (as applicable) involved, the tasks required and shall possess operating knowledge of the particular system or process (as applicable) involved and its relationship to facility safety. The requirements for independent verification are consistent with the applicable guidance provided in ANSI/ANS-3.2, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants.

3.4.6 **Upon completion of the design of IROFS**, the IROFS boundaries will be defined. In defining the boundaries for each IROFS, Louisiana Energy Services procedure "IROFS Boundary Definitions" will be used. This procedure requires the identification of each support system and component necessary to ensure the IROFS is capable of performing its specified safety function.

3.4.7 **The applicable guidance of the following industry standards**, guidance documents and regulatory guides shall be used for the design, procurement, installation, testing, and maintenance of IROFS at the NEF.

a. Institute of Electrical and Electronics Engineers (IEEE) standard IEEE 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
b. IEEE standard 384, "IEEE Standard Criteria for Independence of Class IE Equipment and Circuits"
c. Branch Technical Position HICB-11, "Guidance on Application and Qualification of Isolation Devices," from NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"
d. Regulatory Guide 1.75, "Physical Independence of Electric Systems" e. IEEE standard 344, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"
f. Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants"
g. American National Standards Institute (ANSI)/Instrumentation, Systems, and Automation Society (ISA)-S67.04, Part 1, "Setpoints for Nuclear Safety-Related Instrumentation"

h. Regulatory Guide 3.17, "Earthquake Instrumentation for Fuel Reprocessing Plants," (for IROFS26 only)
i. IEEE standard 338, "IEEE Standard Criteria for Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems"
j. Branch Technical Position HICB-17, "Guidance on Self-Test and Surveillance Test Provisions," from NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"
k. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems"
l. IEEE standard 518, "IEEE Guide for Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources"
m. IEEE standard 1050, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations"
n. IEEE standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations" (for separation and isolation)

3.4.8 **The actual seismic design detailed approach for NEF IROFS** will be based on the DOE-STD-1020 or the ASCE Standard Seismic Design Criteria (ASCE43) method, or in the case of IROFS27e only, on the AISC Manual of Steel Construction and ACI 318. The seismic design will be finalized prior to detailed design.

3.4.9 **To support the final design of the NEF**, additional soil borings were collected from the NEF site. Laboratory testing was performed on soil samples and additional in-situ testing was performed to determine static and dynamic soil properties. Using the soil information obtained, the following activities were conducted.

- The assessment of soil liquefaction potential was performed using the applicable guidance of Regulatory Guide 1.198, Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites.
- Allowable bearing pressures provided in the ISA Summary were refined using the applicable methods of Naval Facilities Engineering Command Design Manual NAVFAC DM-7.02, Foundations and Earth Structures; Foundation Engineering Handbook, H.F. Winterkorn and H.Y. Fang, or Foundation Analysis and Design, J.E. Bowles,.
- Building settlement analysis was performed using the applicable methods of NAVFAC DM-7.01, Soil Mechanics; and Foundation Engineering Handbook, H.F. Winterkorn and H.Y. Fang. The acceptance criteria for the building settlement analysis was based on Urenco design criteria for allowable total and differential settlement of equipment and buildings.

3.4.10 Intentionally Blank

3.4.11 The Separations Building Modules are designed as Type I-B Construction by the NMCBC and as Type II (222) Construction by NFPA 220.

3.4.12 The floors of the Cascade Halls have a floor profile quality classification of flat in accordance with ACI 117 to aid in the transport of assembled centrifuges.

3.4.13 The Technical Services Building is designed as Type II-B Construction by the NMCBC and as Type II (000) Construction by NFPA 220.

3.4.14 The Cylinder Receipt and Dispatch Building is designed as Type I-B Construction by the NMCBC and as Type II (222) Construction by NFPA 220.

3.4.15 The Centrifuge Assembly Building (CAB) is designed as Type II-B Construction by the NMCBC and as Type II (000) Construction by NFPA 220.

3.4.16 As protection of CAB investments (centrifuges and equipment) against the deleterious effects of airborne contaminants, the CAB construction will provide for an ISO 14644-1 Class 8.

3.4.17 The floors of the CAB Assembled Centrifuge Storage Area have a floor profile quality classification of flat in accordance with ACI 117 to aid in the transport of assembled centrifuges.

3.4.18 For QL-1F periodic review of UL and FM recall data UUSA will perform an annual review of UL and FM websites for identification of recall data associated with fire protection basic components.

3.4.19 The Central Utilities Building is designed to meet the occupant and exiting requirements set by the International Fire Code and the New Mexico Commercial Building Code.

3.4.20 The Administration Building is designed to meet the occupant and exiting requirements set by the International Fire Code and the New Mexico Commercial Building Code.

3.4.21 The Central Utilities Building and the Administration Building are designed as Type II-B Construction by the NMCBC and as Type II (000) Construction by NFPA 220.

3.4.22 The following codes and standards are generally applicable to the structural design of the National Enrichment Facility:

- New Mexico Commercial Building Code
- International Building Code
- ASCE 7, Minimum Design Loads for Buildings and Other Structures
- ACI 318, Building Code Requirements for Structural Concrete
- ACI 349, Code Requirements for Nuclear Safety Related Concrete Structures

- AISC Manual of Steel Construction
- ANSI/AISC N690, American National Standard Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities

Historical Note: Fillet weld inspections performed on Cascades 1 and 2 upper steel in SBM1001 Assay Unit 1, under ANSI/AISC N690-1994 and AWS D1. 1, involved use of an alternate weld inspection methodology as approved by the NRC in LAR 11-04. This method, delineated in TQ-2010-102 which has been superseded by TQ-2011-11 to eliminate groove weld applications, involved a through paint weld assessment and engineering evaluation for disposition of identified weld defects.

- PCI Design Handbook
- American Society of Testing and Materials

3.4.23 Structural Design Loads

a. Wind loadings for structures are in accordance with provisions of the International Building Code and Section 6.5 of ASCE 7.

b. For reinforced concrete targets, the formulas used to establish the missile depth of penetration (x) and scabbing thickness (ts) are based on the Modified National Defense Research Committee Formula (NDRC) (ASCE, 58) and the Army Corps of Engineers Formula (ACE) (ASCE, 58) respectively.

c. Per Section C.7.2.1 of ACI 349, the concrete thickness required to resist hard missiles shall be at least 1.2 times the scabbing thickness, ts. Punching shear is calculated and checked against the requirements of ACI 349, Section C.7.2.3.

d. For steel targets, the formula used to establish the perforation thickness is the Ballistic Research Laboratory (BRL) Formula (ASCE, 58).

e. All buildings and structures, including such items as equipment supports, are designed to withstand the earthquake loads defined in Sections 1615 through 1617 of the International Building Code.

f. Extreme snow loadings on roofs of safety significant structures are based on a ground snow load of 32 lb/ft^2. The snow load for safety significant structures is enveloped by the general 40 lb/ft^2 roof live load with the exception of drift areas. Drift areas (where load can exceed 40 lb/ft^2) are evaluated when required for each structure.

   Quality Level 3 structures will as a minimum, meet the IBC requirements for snow loading.

g. Load combinations for concrete structures and components for the safety significant structures are based on ACI 349 except for SBMs which may be based on ACI 318. Load combinations for other concrete structures are based on (ACI 318). All concrete structures are designed using the ACI Strength Design Method (ACI 318).

h. Load combinations for steel structures and components for all buildings are provided in ISAS Section 3.3.2.2.8. All structural steel is designed using the AISC Allowable Stress Method (AISC, Manual of Steel Construction).

i. Design live loads, including impact loads, used are in accordance with Section 4.0 and Table 4-1 of ASCE 7.

j. During detailed design of specific buildings and areas, pressure loads due to postulated truck and pipeline explosions will be considered. The pressure loads will be developed in accordance with the underlying assumptions used in the explosion hazard assessments described in Sections 3.2.1.2.1 and 3.2.2.4 of the ISA Summary. These buildings and areas include: Separations Building Modules (UF_6 Handling Area, Process Services Corridor and Cascade Halls), and the Cylinder Receipt and Dispatch Building Bunkered Area. ISA Summary section 3.3.1, Buildings and Major Components, describes these buildings.

3.4.24 **Natural UF_6 feed** is received at the NEF in Department of Transportation (DOT) 7A, Type A cylinders from a conversion plant. The cylinders are ANSI N14.1, 48Y cylinders. Approximately 20 kg of UF_6 feed material was received at the National Enrichment Facility in ANSI N14.1 30B cylinders to support Hot Acceptance Testing in the CTF.

3.4.25 **Applicable codes and standards** for process systems are reflected in Tables 3.3-1 through 3.3-7.

3.4.26 Product Liquid Sampling Autoclave

a. The pressure vessel is designed and fabricated in accordance with the requirements of ASME Section VIII, Division1, with the exception that the pressure relief devices specified in Sections UG-125 through 137 are not be provided due to the potential for release of hazardous material to the environment through a pressure relief device. Instead, two independent and diverse automatic trips of the autoclave heaters and fan motor are provided to eliminate the heat input and preclude approaching the autoclave design pressure. A large margin exists between the autoclave design pressure 12 bar (174 psia) and the maximum allowable working pressure 1.8 bar (26 psia). The fail-safe design included two independent and diverse automatic trips of the autoclave heaters and fan motor. This meets requirements of ASME Code Case 2211-1 which is listed in ISA Summary Table 3.0-2, Licensing Code Cases of Record. The pressure vessel is also tested and stamped to the requirements of ASME Section VIII, Division 1 rules and is registered with the National Board.
b. The autoclave is designed and tested to ensure leak tight integrity is maintained.
c. The autoclave door seal is leak tested and inspected prior to each autoclave sample sequence.

3.4.27 Pumped Extract GEVS

NOTE: The Heating Ventilation and Air Conditioning (HVAC) systems and Gaseous Effluent Vent Systems (GEVS) for the NEF are undergoing redesign. After these design changes are finalized the information in applicable sections of this report (e.g., 3.4.28 Cylinder Receipt and Dispatch Building, 4.6.1 Ventilation Program, 7.3.5 Ventilation, and 10.1.6 Decommissioning, etc.) will be revised as necessary and in accordance with 10 CRF 70.72. The final design will be evaluated in accordance with the requirements of 10 CFR 70.72 prior to requirements for operational readiness.

a. The Pumped Extract GEVS provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16.

b. The design and in-place testing of the Pumped Extract GEVS will be consistent with the applicable guidance in Regulatory Guide 1.140, ASME AG-1, and ASME N510. The system includes impregnated activated carbon filters for HF removal. As such, the portions of Regulatory Guide 1.140, ASME AG-1, and ASME N510, which address activated carbon filters for radioiodine removal, are not applicable. The prefilter efficiency (60-65%) is based on testing in accordance with ASME AG-1. The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1. The impregnated carbon filter efficiency (99%) for removal of HF is based on Urenco operating experience and specifications. In-place testing and inspections of the HEPA filters will be performed in accordance with the guidance in Regulatory Guide 1.140. The frequency for performance of in-place HEPA filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140. Qualification testing, to verify HF removal efficiency, of the impregnated activated carbon will be performed using ASTM D6646, modified to reflect removal of HF instead of hydrogen sulfide or using an actual in situ test such as described in ETC4044158 (Qualification of Safety by Shape GEVS Filters). Laboratory testing of samples from the impregnated carbon filters will be performed on an annual basis. Throughout the useful life of the impregnated activated carbon, the impregnate is progressively consumed. The laboratory testing will determine the impregnate content within the sample. The amount of impregnate present in the sample is indicative of the remaining life of carbon filter for removal of HF. Carbon filter replacement will be based on the remaining absorption capacity. The remaining filters will be replaced based on differential pressure readings (i.e., filter loading). There is no fixed frequency for filter replacement.

3.4.28 Cylinder Receipt and Dispatch Building (CRDB) GEVS

a. The CRDB GEVS provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16.

b. The design and in-place testing of the CRDB GEVS will be consistent with the applicable guidance in Regulatory Guide 1.140, ASME AG-1, and ASME N510. The system includes an impregnated activated carbon filter for HF removal. As such, the portions of Regulatory Guide 1.140, ASME AG-1, and ASME N510, which address activated carbon filters for radioiodine removal are not applicable. The prefilter efficiency (85%) is based on testing in accordance with ASME AG-1. The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1. The impregnated carbon filter efficiency (99%) for removal of HF is based on Urenco specifications. In-place testing and inspections of the filters will be performed in accordance with the guidance in Regulatory Guide 1.140. The frequency for performance of in-place filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140. Qualification testing, to verify HF removal efficiency, of the impregnated charcoal will be performed using ASTM D6646, modified to reflect removal of HF instead of hydrogen sulfide. Laboratory testing of samples from the impregnated activated carbon filters will be performed on an annual basis. Throughout the useful life of the impregnated carbon, the impregnate is progressively consumed. The laboratory testing will determine the impregnate content within the sample. The amount of impregnate present in the sample is indicative of the remaining life of carbon filter for removal of HF.

3.4.29 Centrifuge Test and Post Mortem Facilities Exhaust Filtration System

The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16.

3.4.30 In response to Bulletin 2003, LES will not purchase UF_6 cylinders with the 1-in Hunt valves installed nor purchase any replacement 1-in valves from Hunt.

In the unlikely event that any cylinders are received at the NEF with the 1-in Hunt valves installed, the following actions will be taken.

- If the cylinder is empty, the valve will be replaced before the cylinder is used in the facility.
- If the cylinder is filled, a safety justification to support continued use of the cylinder until the valve can be replaced will be developed or the valve will be replaced in accordance with NEF procedures.

No cylinders with the 1-in Hunt valve installed will be used as UBCs.

3.4.31 The containers used for intercontinental shipping are International Organization for Standardization Series 1 freight containers that are supplied in accordance with the ISO 668 Standard.

3.4.32 Applicable codes and standards for utility and support systems are reflected in Table 3.3-8.

3.4.33 Exhaust flow from the potentially contaminated rooms (i.e., Ventilated Room and Decontamination Workshop) of the CRDB is filtered by a pre-filter, activated carbon filter and HEPA filter and is then released through an exhaust stack. The exhaust stack flow is continuously monitored for alpha and HF. The stack exhaust is periodically sampled. The continuous monitoring and periodic sampling is in accordance with the guidance in Regulatory Guide 4.16.

3.4.34 The Electrical System design complies with the following codes and standards.

- IEEE C2, National Electrical Safety Code
- New Mexico Electric Code (based on the National Electric Code, NFPA 70)
- NFPA 70E, Standard for Electrical Safety in the Workplace

3.4.35 The criticality safety for tanks that are not "geometrically safe" or "geometrically favorable" will utilize two independent IROFS for mass control, one IROFS is referred to as "bookkeeping measures" and the second IROFS is referred to as "sampled and analyzed," e.g., tank contents are sampled and analyzed before being transferred to another tank or out of the system. The "bookkeeping measures" is a process to calculate the potential mass of uranium in the tank for any batch operation to ensure that no tank holds more than a safe mass of uranium. This calculated mass of uranium is then compared to a mass limit, which is based on the double-batching limit on mass of uranium in a vessel from the criticality safety analyses. The "bookkeeping measures" process is described in further detail below.

- For NEF, the "bookkeeping measures" are only applied to tanks where the mass of uranium involved, even when double batching error is considered, is far below the safe value. Bookkeeping measures are a documented running inventory estimate of the total uranium mass in a particular tank. The mass inventory for each batch operation is calculated based on the mass of material to be transferred during each batch operation and the mass inventory in the tank prior to the addition of the material from the batch operation.
- There are two types of batch operations that are considered. The first type is liquid transfer between tanks based on moving a volume of liquid with uranic material present in the volume. The second is transferring a number of components into the tank with the uranic material contained within or on the components transferred in each batch operation. For both types of operations, the initial mass inventory is set after emptying, cleaning, and readying the tank for receipt of uranic material. For each batch operation, the amount of uranic material to be transferred during a particular batch operation is estimated. This quantity of material is then credited/debited to/from each tank as appropriate. A new mass inventory in each tank is calculated. The calculated receiving tank mass inventory is compared to the mass limit for the tank prior to the transfer.
- For the second type, a transfer of a number of facility components into an open tank during a batch operation, the mass inventory on/within the components is estimated, and that mass credited to the receiving tank. The final mass inventory in the tank is calculated and the total is compared to the mass limit for the tank prior to the transfer. Open tanks associated with this system are located in the Decontamination Workshop.

3.4.36 **UF_6 cylinders with faulty valves** are serviced in the Ventilated Room. In the Ventilated Room, the faulty valve is removed and the threaded connection in the cylinder is inspected. A new valve is then installed in accordance with the requirements of ANSI N-14.1.

3.4.37 **IROFS will be designed**, constructed, tested and maintained to QA Level 1, with the following exceptions,

- IROFS27e which will be designated and analyzed to QA Level 1, and will be constructed, tested, and maintained to QA Level 1 Graded.
- Fire protection features designated as IROFS which will be designed, procured, constructed, tested, and maintained to QA Level 1-Fire Protection (QL-1F)

IROFS will comply with design requirements established by the ISA and the applicable codes and standards (Listed in ISAS Table 3.0-1). IROFS components and their designs will be of proven technology for their intended application. These IROFS components and systems will be qualified to perform their required safety functions under normal and accident conditions for which they are credited, e.g., pressure, temperature, humidity, seismic motion, electromagnetic interference, and radio-frequency interference, as required by the ISA. IROFS components and systems will be qualified using the applicable guidance in Institute of Electrical and Electronics Engineers (IEEE) standard IEEE-323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations". Additionally, non-IROFS components and systems will be qualified to withstand environmental stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of the IROFS safety functions. Furthermore, IROFS components and systems will be designed, procured, installed, tested, and maintained using the applicable guidance in Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems,". IROFS systems will be designed and maintained consistent with the reliability assumptions in the ISA. Redundant IROFS systems will be separate and independent from each other. IROFS systems will be designed to be fail-safe. In addition, IROFS systems will be designed such that process control system failures will not affect the ability of the IROFS systems to perform their required safety functions. Plant control systems will not be used to perform IROFS functions. Installation of IROFS systems will be in accordance with engineering specifications and manufacturer's recommendations. Required testing and calibration of IROFS will be consistent with the assumptions of the ISA and setpoint calculations, as applicable. For hardware IROFS involving instrumentation which provides automatic prevention or mitigation of events, setpoint calculations are performed in accordance with a setpoint methodology, which is consistent with the applicable guidance provided in Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation".

Administrative Control IROFS Support Equipment is defined separately in § 3.4.42

3.4.38 For those IROFS requiring operator actions, a human factors engineering review of the human-system interfaces shall be conducted using the applicable guidance in NUREG-0700, "Human-System Interface Design Review Guidelines,", and NUREG-0711, "Human Factors Engineering Program Review Model."

3.4.39 **LES will review the topography** of the NEF/LES site and surrounding relevant area, out to the boundaries of the drainage basin, for any natural or man made changes. This review will be performed every five years unless significant topography changes are identified between reviews. In the event of changes that could affect the calculation of the maximum possible flood level, LES will re-evaluate the flooding analysis to ensure that all Separations Building Modules (SBMs) abnormal condition calculations are still bounding.

3.4.40 **The Product Stations design will be based on ETC4069917-1** design drawings. The internal station design size of approximately 9'7" does not accommodate a 48-inch feed cylinder. Blending donor and receiver station designs do not accommodate 48-inch cylinders. Product cylinders, as designed, cannot physically connect to a feed station. Therefore, potential for re-feeding enriched materials does not exist. Future construction and design efforts will be consistent. Any modification to station designs or product cylinder connection points will be re-evaluated and revised consistent with overall ISA methodology including criticality reviews.

3.4.41 **The Assay Sampling Rig** shall exhaust to a gaseous effluent ventilation system with safe-by-design attributes. At final design, this rig will be evaluated for criticality concerns and IROFS or other controls will be identified in compliance with 10 CFR 70.61.

3.4.42 Administrative Control IROFS Support Equipment contain attributes that are required by the worker to fulfill the Administrative Control IROFS. The attributes are verified to ensure that the worker can perform the IROFS safety function. Support Equipment is in the Administrative Control IROFS boundary. Many of the actions are to prevent an event and upon failure of indication, actions would be implemented to stop continued operation or not start the operation. However, to enhance worker action and direction to prevent events, Support Equipment was identified and included in the boundary. The attributes of Support Equipment are controlled through the applicable management measures. For example, the attribute of "accurate and reliable indication" is controlled through the calibration and testing which is part of the Maintenance Function Testing Program.

Support Equipment is listed in Table 3.4-1, Administrative Control IROFS Support Equipment. This table contains Support Equipment and other equipment, other equipment is not inside the Administrative Control IROFS boundary; normally such equipment is QL-3. Equipment Attributes are in the Administrative Control IROFS boundary.

Management measures are applied to the attributes of Administrative Control IROFS Support Equipment and other equipment attributes. Management measures are also applied to Administrative Control IROFS Support Equipment as defined in the Quality Assurance Program Description for QL-2AC equipment.

3.5 References

Edition of Codes, Standards, NRC Documents, etc that are not listed below are given in ISAS Table 3.0-1.

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.65, Additional content of applications, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.62, Safety program and integrated safety analysis, 2003.

CFR, 2003e. Title 29, Code of Federal Regulations, Section 1910, Occupational Safety and Health Standards, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Section 70.72, Facility changes and change process, 2003.

LES, 1993. Claiborne Enrichment Center Safety Analysis Report, Louisiana Energy Services, December 1993.

3.6 Chapter 3 Tables

Table 3.1-1 HAZOP Guidewords

UF_6 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 UF_6 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	
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	

3.6 Chapter 3 Tables

Table 3.1-2 ISA HAZOP Table Sample Format

ISA HAZOP NODE:		DESCRIPTION :			DATE:	PAGE:
GUIDEWORD	HAZARD	CAUSE	CONSEQUENCE	SAFEGUARDS	MITIGATING FACTORS	COMMENTS

3.6 Chapter 3 Tables

Table 3.1-3 Consequence Severity Categories Based on 10 CFR 70.61

	Workers	Offsite Public	Environment
Category 3 High Consequence	Radiation Dose (RD) >1 Sievert (Sv) (100 rem) Chemical Dose (CD) > AEGL-3 for HF CD> AEGL-3 for U	RD > 0.25 Sv (25 rem) 30 mg sol U intake CD > AEGL-2	–
Category 2 Intermediate Consequence	0.25 Sv (25 rem) <RD≤ 1 Sv (100 rem) AEGL-2 < CD≤ AEGL-3 for HF AEGL-2< CD≤ AEGL for U	0.05 Sv (5 rem) < RD≤ 0.25 Sv (25 rem) AEGL-1 <CD≤ AEGL-2	Radioactive release > 5000 x Table 2 Appendix B of 10 CFR Part 20
Category 1 Low Consequence	Accidents of lower radiological and chemical exposures than those above in this column	Accidents of lower radiological and chemical exposures than those above in this column	Radioactive releases with lower effects than those referenced above in this column

Notes:

* The worker that causes the release is expected to immediately sense and recognize the release and will not receive a dose significantly greater than a worker elsewhere in the room.

3.6 Chapter 3 Tables

Table 3.1-4 Chemical Dose Information

	High Consequence (Category 3)	**Intermediate Consequence (Category 2)**
Worker	> 146 mg U/m^3 > 139 mg HF/m^3	> 19 mg U/m^3 > 78 mg HF/m^3
Public (outside controlled area) (30-min exposure)	> 13 mg U/m^3 > 28 mg HF/m^3	> 2.4 mg U/m^3 > 0.8 mg HF/m^3

Table 3.1-5 Likelihood Categories Based on 10 CFR 70.61

	Likelihood Category	**Probability of Occurrence***
Not Unlikely	3	More than 10^{-4} per-event per-year
Unlikely	2	Between 10^{-4} and 10^{-5} per-event per-year
Highly Unlikely	1	Less than 10^{-5} per-event per-year

*Based on approximate order-of-magnitude ranges

3.6 Chapter 3 Tables

Table 3.1-6 Risk Matrix with Risk Index Values

Severity of Consequences	Likelihood of Occurrence		
	Likelihood Category 1 Highly Unlikely (1)	Likelihood Category 2 Unlikely (2)	Likelihood Category 3 Not Unlikely (3)
Consequence Category 3 High (3)	Acceptable Risk 3	Unacceptable Risk 6	Unacceptable Risk 9
Consequence Category 2 Intermediate (2)	Acceptable Risk 2	Acceptable Risk 4	Unacceptable Risk 6
Consequence Category 1 Low (1)	Acceptable Risk 1	Acceptable Risk 2	Acceptable Risk 3

Table 3.1-7 (Not Used)

Table 3.1-8 Determination of Likelihood Category

Likelihood Category	Likelihood Index T (= sum of index numbers)
1	$T \leq -5$
2	$-5 < T \leq -4$
3	$-4 < T$

3.6 Chapter 3 Tables

Table 3.1-9 Failure Frequency Index Numbers

Frequency Index No.	Based On Evidence	Based On Type Of IROFS**	Comments
-6*	External event with freq. < 10^{-6} /yr		If initiating event, no IROFS needed.
-5*	Initiating event with freq. < 10^{-5}/yr		For passive safe-by-design components or systems, failure is considered highly unlikely when no potential failure mode (e.g., bulging, corrosion, or leakage) exists, as discussed in Section 3.1.3.2, significant margin exists*** and these components and systems have been placed under configuration management.
-4*	No failures in 30 years for hundreds of similar IROFS in industry	Exceptionally robust passive engineered IROFS (PEC), or an inherently safe process, or two independent active engineered IROFS (AECs), PECs, or enhanced admin. IROFS	Rarely can be justified by evidence. Further, most types of single IROFS have been observed to fail
-3*	No failures in 30 years for tens of similar IROFS in industry	A single IROFS with redundant parts, each a PEC or AEC	
-2*	No failure of this type in this facility in 30 years	A single PEC	
-1*	A few failures may occur during facility lifetime	A single AEC, an enhanced admin. IROFS, an admin. IROFS with large margin, or a redundant admin. IROFS	
0	Failures occur every 1 to 3 years	A single administrative IROFS	
1	Several occurrences per year	Frequent event, inadequate IROFS	Not for IROFS, just initiating events
2	Occurs every week or more often	Very frequent event, inadequate IROFS	Not for IROFS, just initiating events

*Indices less than (more negative than) –1 should not be assigned to IROFS unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the IROFS may be changed or not maintained.

**The index value assigned to an IROFS of a given type in column 3 may be one value higher or lower than the value given in column 1. Criteria justifying assignment of the lower (more negative) value should be given in the narrative describing ISA methods. Exceptions require individual justification.

***For components that are safe-by-volume, safe-by-diameter, or safe-by-slab thickness, significant margin is defined as a margin of at least 10%, during both normal and upset conditions, between the actual design parameter value of the component and the value of the critical design attribute. For components that require a more detailed criticality analysis, significant margin is defined as $k_{eff} < 0.95$, where $k_{eff} = k_{calc} + 3\sigma_{calc}$.

3.6 Chapter 3 Tables

Table 3.1-10 Failure Probability Index Numbers

Probability Index No.	Probability of Failure on Demand	Based on Type of IROFS	Comments
-6*	10^{-6}		If initiating event, no IROFS needed.
-4 or -5*	10^{-4} - 10^{-5}	Exceptionally robust passive engineered IROFS (PEC), or an inherently safe process, or two redundant IROFS more robust than simple admin. IROFS (AEC, PEC, or enhanced admin.)	Can rarely be justified by evidence. Most types of single IROFS have been observed to fail
-3 or -4*	10^{-3} - 10^{-4}	A single passive engineered IROFS (PEC) or an active engineered IROFS (AEC) with high availability	
-2 or -3*	10^{-2} - 10^{-3}	A single active engineered IROFS, or an enhanced admin. IROFS, or an admin. IROFS for routine planned operations	
-1 or -2	10^{-1} - 10^{-2}	An admin. IROFS that must be performed in response to a rare unplanned demand	

*Indices less than (more negative than) –1 should not be assigned to IROFS unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the IROFS may be changed or not maintained.

Table 3.1-11 Failure Duration Index Numbers

Duration Index No.	Avg. Failure Duration	Duration in Years	Comments
1	More than 3 yrs	10	
0	1 yr	1	
-1	1 mo	0.1	Formal monitoring to justify indices less than -1
-2	A few days	0.01	
-3	8 hrs	0.001	
-4	1 hr	10^{-4}	
-5	5 min	10^{-5}	

3.6 Chapter 3 Tables

Table 3.3-1 Cascade System Codes and Standards

The Centrifuge Machine Passive Isolation Devices is designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards.
All process piping in the Cascade System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3.
The design of electrical systems and components in the Cascade System is in conformance with the requirements of the National Electrical Safety Code, IEEE C2, and New Mexico Electric Code (based on the National Electric Code, NFPA 70), and appropriate industry codes and standards.

Editions of Codes, Standards, NRC Documents, etc are listed in ISAS Table 3.0-1.

Table 3.3-2 Product Take-off System Codes and Standards

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Product Take-off System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Product Take-off System.
Material handling equipment is designed in accordance with the appropriate industry codes and standards and the requirements of the Occupational Safety and Health Administration. There is no QA Level 1 material handling equipment in the Product Take-off System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Product Take-off System.
All process piping in the Product Take-off System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3.
All 30-in cylinders used in the Product Take-off System comply with the requirements of ANSI N14.1, Uranium Hexafluoride Packaging for Transport.

Editions of Codes, Standards, NRC Documents, etc are listed in ISAS Table 3.0-1.

3.6 Chapter 3 Tables

Table 3.3-3 Tails Take-off System Codes and Standards

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Tails Take-off System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Tails Take-off System.
Material handling equipment is designed in accordance with the appropriate industry codes and standards and the requirements of the Occupational Safety and Health Administration. There is no QA Level 1 material handling equipment in the Tails Take-off System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Tails Take-off System.
All process piping in the Tails Take-off System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3.
All 48-in cylinders used in the Tails Take-off System comply with the requirements of ANSI N14.1, Uranium Hexafluoride Packaging for Transport.

Editions of Codes, Standards, NRC Documents, etc are listed in ISAS Table 3.0-1.

Table 3.3-4 Product Blending System Codes and Standards

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Product Blending System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Product Blending System.
Material handling equipment is designed in accordance with the appropriate industry codes and standards and the requirements of the Occupational Safety and Health Administration. There is no QA Level 1 material handling equipment in the Product Blending System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Product Blending System.
All process piping in the Product Blending System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3.
All 30-in cylinders used in the Product Blending System comply with the requirements of ANSI N14.1, Uranium Hexafluoride Packaging for Transport.

Editions of Codes, Standards, NRC Documents, etc are listed in ISAS Table 3.0-1.

3.6 Chapter 3 Tables

Table 3.3-5 Product Liquid Sampling System Codes and Standards

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Product Liquid Sampling Autoclaves and their supports are designed to meet the requirements of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section VIII, Division I.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Product Liquid Sampling System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Product Liquid Sampling System.
Material handling equipment is designed in accordance with the appropriate industry codes and standards and the requirements of the Occupational Safety and Health Administration. There is no QA Level 1 material handling equipment in the Product Liquid Sampling System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Product Liquid Sampling System.
All process piping in the Product Liquid Sampling System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3.
All 1.5-in and 30-in cylinders used in the Product Liquid Sampling System comply with the requirements of ANSI N14.1, Uranium Hexafluoride Packaging for Transport.

Editions of Codes, Standards, NRC Documents, etc are listed in ISAS Table 3.0-1.

Table 3.3-6 Contingency Dump System Codes and Standards

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Contingency Dump System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Contingency Dump System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Contingency Dump System.
All process piping in the Contingency Dump System meets or exceeds the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3.

Editions of Codes, Standards, NRC Documents, etc are listed in ISAS Table 3.0-1.

3.6 Chapter 3 Tables

Table 3.3-7 GEVS Codes and Standards

Equipment Type	Code or Standard
Air Handling Units	NFPA 90A AMCA Pub. 99 AMCA Pub. 261 ARI 430 NEMA MG 1
Fans/Motors	AMCA 210 ASHRAE 51 ASHRAE Systems and Equipment NEMA MG1
Coils	ANSI/ARI 410
Air Cleaning Devices	ASME AG-1 ERDA 76-21 ANSI/ASME N509 ANSI/ASME N510 ASTM D6646 (See Note 1) ANSI/AWS-D1-1.1 (for Pumped Extract GEVS) ANSI/AWS-D1.3 (for Pumped Extract GEVS) ANSI/AWS-D9.1 (for CRDB GEVS)
Dampers	

Editions of Codes, Standards, NRC Documents, etc are listed in ISAS Table 3.0-1.

Note 1. Qualification testing, to verify HF removal efficiency, of the impregnated carbon will be performed using ASTM D6646, modified to reflect removal of HF instead of hydrogen sulfide or using an actual in situ test such as described in ETC4044158 (Qualification of Safe by Shape GEVS Filters).

3.6 Chapter 3 Tables

Table 3.3-8 Utility and Support Systems Codes and Standards

ACI 318, Building Code Requirements for Structural Concrete.
ACI 349, Code Requirements for Nuclear Safety Related Concrete Structures.
AIChE, Guidelines for Hazard Evaluation Procedures.
AISC Manual of Steel Construction – Allowable Stress Design
ANSI N14.1, American National Standard for Nuclear Materials - Uranium Hexafluoride Packaging for Transport.
ANSI N15.5, Statistical Terminology and Notation for Nuclear Materials Management.
ASCE 58, Structural Analysis and Design of Nuclear Plant Facilities, Manuals and Reports on Engineering Practice.
ASCE 7, Minimum Design Loads for Building and Other Structures.
ASME B31.3, Process Piping.
ASME, Boiler and Pressure Vessel Code, Section VIII, Division 1.
ASME, NQA-1, Quality Assurance Requirements for Nuclear Facility Applications.
ASTM C761 - Standard Test Methods for Chemical, Mass Spectrometric, Spectrochemical, Nuclear, and Radiochemical Analysis of Uranium Hexafluoride.
ASTM E 814, Fire Tests of Through-Penetration Fire Stops.
ERDA 76-21, Nuclear Air Cleaning Handbook.
IEEE 336, Standard Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities.
IEEE C2, National Electrical Safety Code.
IFC, International Fire Code
ISO 668, Series 1 Freight Containers - Classification, Dimensions and Ratings.
NFPA 1, Fire Prevention Code.
NFPA 10, Portable Fire Extinguishers.
NFPA 12, Carbon Dioxide Systems.
NFPA 13, Installation of Sprinkler Systems.
NFPA 14, Standpipe, Private Hydrant and Hose Systems.
NFPA 15, Water Spray Fixed Systems for Fire Protection.
NFPA 20, Installation of Stationary Pumps.
NFPA 2001, Clean Agent Fire Extinguishing Systems.
NFPA 22, Water Tanks for Private Fire Protection.
NFPA 221, Fire Walls and Fire Barrier Walls.
NFPA 24, Private Fire Service Mains and Their Appurtenances.
NFPA 25, Water Based Fire Protection Systems.
NFPA 30, Flammable and Combustible Liquids Code.

3.6 Chapter 3 Tables

Table 3.3-8 Utility and Support Systems Codes and Standards

NFPA 5000, Building Construction and Safety Code.
NFPA 54, National Fuel Gas Code.
NFPA 55, Compressed & Liquefied Gases in Cylinders.
NFPA 58, Liquefied Petroleum Gas Code.
NFPA 600 Industrial Fire Brigades.
New Mexico Electric Code (based on the National Electric Code, NFPA 70)
NFPA 704, Standard System for the Identification of the Hazards of Materials for Emergency Response.
NFPA 72, National Fire Alarm Code.
NFPA 75, Electronic Computer/Data Processing Systems.
NFPA 780, Lightning Protection Systems.
NFPA 80, Fire Doors and Fire Windows.
NFPA 801, Fire Protection for Facilities Handling Radioactive Materials.
NFPA 80A, Exterior Fire Exposures.
NFPA 90A, Installation of Air Conditioning and Ventilating Systems.
NFPA 90B, Installation of Warm Air Heating and Air Conditioning Systems.
NFPA 91, Exhaust Systems for Air Conveying of Materials.
NFPA, Fire Protection Handbook, Section 9, Chapter 30, Nuclear Facilities.
NFPA 110, Standard for Emergency and Standby Power Systems.
NFPA 111, Standard on Stored Electrical Energy Emergency and Standby Power Systems.
NFPA 70E, Standard for Electrical Safety in the Workplace.
NFPA 79, Electrical Standard for Industrial Machinery.
PCI Design Handbook.
International Building Code (as amended by the NMCBC).
Uniform Mechanical Code (as amended by the New Mexico Mechanical Code).
Uniform Plumbing Code (as amended by the New Mexico Plumbing Code).

Editions of Codes, Standards, NRC Documents, etc are listed in Table 3.0-1.

3.6 Chapter 3 Tables

Table 3.4-1 Administrative Control IROFS Support Equipment

IROFS	Monitoring Support Equipment	Other Equipment	Equipment Attributes	Operated Support Equipment	Other Equipment	Equipment Attributes
IROFS14a	None	Two independent instruments for determining gross ^{235}U content	Accurate and reliable indication	None	None	None
IROFS14b	None	Two independent instruments for determining gross ^{235}U content	Accurate and reliable indication	None	None	None
IROFS16a	None	Instrument for viewing cylinder internal	None	None	None	None
	None ***(Note 1)**	M&TE Instrument ***(Note 1)**	Accurate and reliable indication	None	None	None
	Pressure instrument ***(Note 2)**	None	Accurate and reliable indication	None	None	None
IROFS30a	None	None	None	None	None	None
IROFS30b	None	Oil analyzer	Accurate and reliable indication	None	None	None
IROFS30c	None	Oil analyzer	Accurate and reliable indication	None	None	None
IROFS31a	None	Instrument for determining gross ^{235}U content, independent of IROFS31b	Accurate and reliable indication	None	None	None
IROFS31b	None	Instrument for determining gross ^{235}U content, independent of IROFS31a	Accurate and reliable indication	None	None	None

3.6 Chapter 3 Tables

Table 3.4-1 Administrative Control IROFS Support Equipment						
IROFS	**Monitoring Support Equipment**	**Other Equipment**	**Equipment Attributes**	**Operated Support Equipment**	**Other Equipment**	**Equipment Attributes**
IROFS31c	None	None	None	None	None	None
IROFS36c	None	Fuel Tank	Fuel Tank Volume	None	None	None
IROFS36e	None	Fuel Tank	Fuel Tank Volume	None	None	None
	None	UBC Storage Pad Slope	Slope of the Pad to prevent excess pooling	None	None	None
IROFS36f	None	Topographical survey equipment	Accurate and reliable topography reading	None	None	None
IROFS36g	None	None	None	None	Landscape Equipment	None
IROFS38	Weighing Scale System including local digital readout from weighing system at the cylinder stations ***(Notes 2 and 3)**	None	Accurate and reliable indication	Select independent isolation valves ***(Note2)**	None	Valve closure
IROFS39a	None	None	None	None	None	None
IROFS39b	None	None	None	None	None	None
IROFS39c	None	None	None	None	None	None
IROFS39d	None	None	None	None	None	None
IROFS42	Product Station Weighing Scale System including local digital readout from weighing system at the	None	Accurate and reliable indication	None	None	None

Table 3.4-1 Administrative Control IROFS Support Equipment

IROFS	Monitoring Support Equipment	Other Equipment	Equipment Attributes	Operated Support Equipment	Other Equipment	Equipment Attributes
	product stations ***(Notes 2 and 3)**					
IROFS50a	None	None	None	None	Barriers	Visible and substantial
IROFS50b	None	None	None	None	Barriers	Visible and substantial
IROFS50c	None	None	None	None	Barriers	Visible and substantial
IROFS50d	None	None	None	None	Barriers	Visible
IROFS50e	None	None	None	None	None	None
IROFS50f	None	None	None	None	Barriers	Visible and substantial
IROFS50g	None	None	None	None	Barriers	Visible
IROFS50h	None	None	None	None	Barriers	Visible and substantial
IROFSC22	1) Weigh Scale System including local digital readout from weighing system at cylinder station ***(Notes 2 and 3)**	None	1) Accurate and reliable indication	Select independent isolation valves ***(Note 2)**	None	Valve closure
	2) vent system cold trap load cells ***(Notes 2 and 3)**		2) Accurate and reliable indication			

***(Note 1)** M&TE will be used for Initial Plant Operations until the in-line process instrumentation is installed. The M&TE is QA Level 3 equipment calibrated in accordance with the Maintenance Management Measure. The permanently installed pressure instrument will meet the requirements for QA Level 2AC.

***(Note 2)** Support Equipment meets the requirements for QA Level 2AC.

*(Note 3) An exception to License Condition 20 has been approved by the NRC for this equipment.

4.0 Radiation Protection

This chapter describes the facility Radiation Protection Program. The Radiation Protection Program protects the radiological health and safety of workers and complies with the regulatory requirements in 10 CFR 19 (CFR, 2003a), 20 (CFR, 2003b) and 70 (CFR, 2003c).

The information provided in this chapter, the corresponding regulatory requirement and the NRC acceptance criteria from NUREG-1520, Chapter 4 is summarized in the table below. Information beyond that required by the Standard Review Plan is included.

Information Category and Requirement	**10 CFR Citation**	**NUREG-1520 Chapter 4 Reference**
Section 4.1 Commitment to Radiation Protection Program Implementation	10 CFR 20.1101, Subpart B	4.4.1.3
Section 4.2 Commitment to an ALARA Program	10 CFR 20.1101	4.4.2.3
Section 4.3 Organization and Personnel Qualifications	10 CFR 70.22	4.4.3.3
Section 4.4 Commitment to Written Procedures	10 CFR 70.22(a)(8)	4.4.4.3
Section 4.5 Training Commitments	10 CFR 19.12 & 10 CFR 20.2110	4.4.5.3
Section 4.6 Ventilation and Respiratory Protection Programs Commitments	10 CFR 20, Subpart H	4.4.6.3
Section 4.7 Radiation Surveys and Monitoring Programs Commitments	10 CFR 20, Subparts F, C, L, M	4.4.7.3
Section 4.8 Contamination and Radiation Control	N/A	N/A
Section 4.9 Maintenance Areas - Methods and Procedures for Contamination Control	N/A	N/A
Section 4.10 Decontamination Policy and Provisions	N/A	N/A
Section 4.11 Additional Program Commitments	N/A	4.4.8.3

4.1 Commitment to Radiation Protection Program Implementation

The radiation program meets the requirements of 10 CFR 20 (CFR, 2003b), Subpart B, Radiation Protection Programs, and is consistent with the guidance provided in Regulatory Guide 8.2, Guide for Administrative Practice in Radiation Monitoring. The facility develops, documents and implements its Radiation Protection Program commensurate with the risks posed by a uranium enrichment operation. The facility uses, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA). The radiation program content and implementation are reviewed at least annually as required by 10 CFR 20.1101(c) (CFR, 2003d). In addition, in accordance with 10 CFR 20.1101(d) (CFR, 2003d) constraints on atmospheric releases are established for the NEF such that no member of the public would be expected to receive a total effective dose equivalent in excess of 0.1 mSv/yr (10 mrem/yr) from these releases. Additional information regarding compliance with 10 CFR 20.1101(d) is provided in Section 9.2.

Protection of plant personnel requires (a) surveillance of and control over the radiation exposure of personnel; and (b) maintaining the exposure of all personnel not only within permissible limits, but "as low as is reasonably achievable," in compliance with applicable regulations and license conditions. The objectives of Radiation Protection are to prevent acute radiation injuries (nonstochastic or deterministic effects) and to limit the potential risks of probabilistic (stochastic) effects (which may result from chronic occupational exposure) to an acceptable level.

The facility's philosophy for radiation protection is reflected in the establishment of a Radiation Protection Program that has the specific purpose of maintaining occupational radiation exposures ALARA. The program includes written procedures, periodic assessments of work practices and internal/external doses received, work plans and the personnel and equipment required to ensure implementation of ALARA goal.

The facility's administrative personnel exposure limits are set below the limits specified in 10 CFR 20 (CFR, 2003b) to provide assurance that legal radiation exposure limits are not exceeded and that the ALARA principle is emphasized. The facility administrative exposure limits are given in Table 4.1-1, Administrative Radiation Exposure Limits. Estimates of the facility area radiation dose rates and individual personnel exposures, during normal operations, are shown in Table 4.1-2, Estimated Dose Rates and Table 4.1-3, Estimated Individual Exposures. These estimates are based upon the operating experience of similar Urenco facilities in Europe.

The annual dose equivalent accrued by a typical radiation worker at a uranium enrichment plant is low. At the Urenco Capenhurst plant, the maximum annual worker dose equivalent was 3.1 mSv (310 mrem), 2.2 mSv (220 mrem), 2.8 mSv (280 mrem), 2.7 mSv (270 mrem) and 2.3 mSv (230 mrem) during the years 1998 through 2002, respectively. For each of these same years, the average annual worker dose equivalent was approximately 0.2 mSv (20 mrem) (Urenco, 2000; Urenco, 2001; Urenco, 2002).

The radiation exposure policy and control measures for personnel are set up in accordance with requirements of 10 CFR 20 (CFR, 2003b) and the guidance of applicable Regulatory Guides. Recommendations from the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP) may also be used in the formulation and evolution of the facility Radiation Protection Program.

The facility corrective action process is implemented if (1) personnel dose monitoring results 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.

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. The compliance responsibilities are activities associated with nuclear safety and monitoring radioactive effluents. .

4.1.1.3 Environmental Compliance Officer

The Environmental Compliance Officer reports to the Health, Safety, and Environmental Manger and has the responsibility for coordination facility activities to ensure all local, state and federal environmental regulations are met.

4.1.1.4 Radiation Protection Manager

The Radiation Protection Manager reports to the Director of Compliance 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 is responsible for:

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

- 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 as related to calibration and performance check sources
- Supports the Recycling Manager in the 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
- Posting 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.

4.1.1.5 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.

4.1.1.6 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 accordance with the Emergency Plan. Staffing of the Radiation Protection Program consists of the Radiation Protection Manager and Radiation Protection Program staff members who are radiation protection technician qualified. In addition, there are task qualified personnel outside of the Radiation Protection Department to handle routine radiation protection functions as necessary, and to provide additional response capability in an emergency. The radiation protection technician staffing level is reassessed as the workload and plant expands.

4.1.3 Independence of the Radiation Protection Program

The Radiation Protection Program is independent of the facility's routine operations. This independence ensures that the Radiation Protection Program maintains its objectivity and is focused only on implementing sound radiation protection principles necessary to achieve occupational doses and doses to members of the public that are ALARA. As previously noted in Section 4.1.1.3, Radiation Protection Manager, that in matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager.

4.1.4 Radiation Safety Committee

The Radiation Safety Committee meets periodically to review, in accordance with 10 CFR 20.1101(c) (CFR, 2003d), the status of projects, measure performance, look for trends and to review radiation safety aspects of facility operations. The Radiation Protection Manager chairs the Radiation Safety Committee. Radiation Safety Committee members are from quality assurance, operations, maintenance, and technical support, as deemed appropriate by the Plant Manager.

The objectives of the Radiation Safety Committee are to maintain a high standard of radiation protection in all facility operations. The Radiation Safety Committee reviews the content and implementation of the Radiation Protection Program at a working level and strives to improve the program by reviewing exposure trends, the results of audits, regulatory inspections, worker suggestions, survey results, exposure incidents, etc.

The maximum interval between meetings may not exceed 180 days. A written report of each Radiation Safety Committee meeting is forwarded to all Managers.

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, RCAs 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.

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.

4.2 Commitment to an ALARA Program

Programs for improving the effectiveness of equipment used for effluent and exposure control are evaluated by the SRC and recommendations are documented in writing. The implementation of the committee's recommendations is tracked to completion via the Corrective Action Program, which is described in Section 11.6, Incident Investigations and Correction Action Process.

The SRC also reviews the effectiveness of the ALARA program and determines if exposures, releases and contamination levels are in accordance with the ALARA concept. It also evaluates the results of assessments made by the radiation protection organization, reports of facility radiation levels, contamination levels, and employee exposures for identified categories of workers and types of operations. The committee is responsible for ensuring that the occupational radiation exposure dose limits of 10 CFR 20 (CFR, 2003b) are not exceeded under normal operations. The committee determines if there are any upward trends in personnel exposures, environmental releases and facility contamination levels.

The ALARA program facilitates interaction between radiation protection and operations personnel by being comprised staff members from those organizations. The SRC periodically reviews the goals and objectives of the ALARA program and incorporates, as appropriate, new technologies or approaches and operating procedures or changes that could cost-effectively reduce potential radiation exposures.

4.3 Organization and Personnel Qualifications

The regulation 10 CFR 70.22 (CFR, 2003h) requires that the technical qualifications, including training and experience of facility staff be provided in the license application. This information is provided in this section.

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

The Radiation Protection Manager's qualification requirements are described in Section 2.2.4. As stated in Section 4.1.2, Staffing of the Radiation Protection Program, other 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.

The Radiation Protection Manager reports to the Director of Compliance and has the responsibility for establishing and implementing the Radiation Protection Program. Duties include training of personnel in use of equipment, control of radiation exposure of personnel, continuous determination and evaluation of the radiological status of the facility. The radiological environmental monitoring program is a function of the Environmental Compliance Officer. The facility organization chart establishes clear organizational relationships among the radiation protection staff and the other facility line managers. The facility operating organization is described in Chapter 2, Organization and Administration.

In all matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager. The Radiation Protection Manager is skilled in the interpretation of radiation protection data and regulations. The Radiation Protection Manager is also familiar with the operation of the facility and radiation protection concerns relevant to the facility. The Radiation Protection Manager is a resource for radiation safety management decisions.

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 in the License Basis Documents.

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

All work performed in a Radiologically Controlled Area (RCA) is performed in accordance with a Radiation Work Permit (RWP). The RWP provides a description of the 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 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.

RWPs are issued for routinely performed activities for extended durations in areas where radiological conditions are well characterized and not expected to change, such as tours of the plant by shift personnel or the changing of cylinders. A new RWP is not issued for routine activities where radiological conditions are not expected to change.

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 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
- RWPs include requirements for any necessary radiological safety controls, personnel monitoring devices, protective clothing, respiratory protective equipment, air sampling equipment and the attendance of radiation protection technicians at the work location
- 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 until termination of the license requiring the record in compliance with 10 CFR 20.2103 (CFR, 2003v).

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.

Personnel entering the Radiologically Controlled Areas (RCAs) receive training that is commensurate with the radiological hazard to which they may be exposed.

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).

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.

4.5 Training Commitments

Continuing Training of personnel with RCA access 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.

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. Task specific 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 are escorted by trained personnel while in an RCA. Visitors to the RCA receive a radiological briefing commensurate with their entry in accordance with 10 CFR 19.12.

Personnel access procedures ensure the completion of nuclear safety worker training prior to permitting unescorted access into 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.

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 of the training. The effectiveness of the training programs is evaluated by audits and assessments

I 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 staff.

Content 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.

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. 127-Operational Radiation Safety Program
- Regulatory Guide 8.15-Acceptable Programs for Respiratory Protection

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 UF_6 within process equipment. The entire UF_6 enrichment process, except for liquid sampling, is operated under a partial vacuum so that leaks are into the system and not into work areas.

Building ventilation systems control the temperature and the humidity of the air inside the building. Note: Not all buildings will have humidity control. 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. 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.

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.

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 practical 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:

A. Control of access
B. Limitation of exposure times
C. Use of respiratory protection equipment
D. Other controls, as available and appropriate.

If an ALARA analysis is performed to determine whether or not respirators should be used, safety factors other than radiological factors may be considered. The impact of respirator use on workers' industrial health and safety is factored into decisions to use respirators.

When respiratory protection equipment is used to limit the intake of radioactive material, only National Institute of Occupational Safety and Health (NIOSH) certified equipment is used. The respiratory protection program meets the requirements of 10 CFR 20 (CFR, 2003b), Subpart H (Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas).

The respiratory protection program includes the following elements:

4.6 Ventilation and Respiratory Protection Programs Commitments

A. Air sampling to identify the potential hazard, select proper equipment and estimate doses

B. Surveys and, when necessary, bioassays to evaluate actual intakes

C. Performance testing of respirators for operability (user seal check for face sealing devices and functional check for others) immediately prior to each use.

D. Written procedures for the following:

1. Monitoring, including air sampling and bioassays
2. Supervision and training of respirator users
3. Fit testing
4. Respirator selection
5. Breathing air quality
6. Inventory and control
7. Storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment
8. Record keeping
9. Limitations on periods of respirator use and relief from respirator use.

E. Determination by a physician, or designee working under the physician's license, that the individual user is medically fit to use respiratory protection equipment:

1. Before the initial fitting of a face sealing respirator
2. Before the first field use of non-face sealing respirators
3. Either every 12 months thereafter, or periodically at a frequency determined by a physician.

F. A respirator fit test requires a minimum fit factor of at least 10 times the Assigned Protection Factor (APF) for negative pressure devices, and a fit factor of at least 500 for any positive pressure, continuous flow, and pressure-demand devices. The fit testing is 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, 2003l)).
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.

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
- ANSI/HPS N13.30-Performance Criteria for Radiobioassay
- ANSI N13.6, Practice for Occupational Radiation Exposure Records Systems

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 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 require individual monitoring of external occupational dose per 10 CFR 20.1502(a) 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 an RCA where the potential for airborne radioactivity or surface contamination may exist.

Continuous airborne radioactivity monitors provide indication of the airborne activity levels in 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.

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 typically needed 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 UF_6 Feed, and Figure 4.7-1, Uranium and Decay Products of Interest, respectively. However, electronic dosimeters may be prescribed for specific work evolutions, as warning devices in areas where dose rates may vary or as alternate dosimetry for visitors or workers who are escorted into the RCA and do not require monitoring per 10 CFR 20.1502(a).

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.

4.7.1 Radiological Areas

Radiological 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 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. The Restricted Area is depicted in Figure 4.7-2 Projected Radiological Zones.

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. Contamination monitoring equipment is located at RCA egress points where loose contamination exists. Personnel who have not been trained in radiological hazard awareness and radiological work practices are not allowed access to a RCA without a trained escort.

The areas defined below may exist within an RCA. 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).
- LES defines a "Contaminated Area" as an area where removable contamination levels are above 16.7 Bq/100 cm^2 (1,000 dpm/100 cm^2) 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. Areas where the intake of soluble uranium in one week is likely to exceed 1 milligram are posted.

4.7.1.4 Controlled Area

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. Individuals who are contractor or facility employees and who work only in the Controlled Area are occupationally exposed but typically exclude from individual monitoring requirements under 10 CFR 20.1502.

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) 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.

Access to and egress from an RCA is through a local control point. 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 an RCA and any additional designated areas within an RCA (e.g., a Contaminated Area which is provided with a step-off pad and frisker) shall not exceed 16.7 Bq/100 cm^2 (1,000 dpm/100 cm^2) 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.

4.7.3 Posting for Radiological Hazardous Awareness

Radiological hazard awareness training is provided through a General Employee Training program. Radiological hazards are identified throughout the Restricted Area and barriers, postings, and labeling per the requirements of 10 CFR 20, Subpart J (CFR, 2003q) are established. Radioactive material storage locations are posted "Caution Radioactive Material". Radioactive material is transit between storage locations is attended by an individual to control the radiological hazard and radioactive material.

4.7.4 Protective Clothing and Equipment

The proper use of protective clothing and respiratory protection 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 as prescribed by the RWP. 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.

Radiation protection management and associated technical staff are responsible for determining the need for protective clothing in each work area.

4.7.5 Personnel Monitoring for External Exposures

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). Personnel whose duties routinely require them to enter an RCA wear individual external dosimetry devices, e.g., 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 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.

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 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. Air monitoring in Airborne Radioactivity Areas is performed as necessary to supplement the bioassay program. Alarm setpoints on the air monitors in 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.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.

4.7.7 Local Control Points

Monitor stations Local Control Points are the entry and exit points for RCAs where loose surface contamination is likely to exist. Monitors are provided, as required, 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. Radiation protection management is responsible for Local Control Point provision and maintenance. Local Control Point locations are evaluated and established as necessary in response to changes in the facility radiological conditions.

4.7.8 Personnel Decontamination

- 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.

4.7.9 Storage Areas

Storage areas are provided for the following items:

- Protective (i.e., anti-contamination) clothing
- Respiratory protection equipment
- Personnel Decontamination 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 UF_6 precludes direct contact with radioactive materials by personnel.
- Self-monitoring is required upon exit from 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.

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.

4.8.1.1 Bioassay

Internal radiological exposures are evaluated as noted in Section 4.7.6, Personnel Monitoring for Internal Exposures. 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 ^{234}U, 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 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:

- Weekly and following any indication of release that might lead to airborne concentrations of uranium that are likely to exceed (1) 1 Derived Air Concentration (DAC) (most likely uranium isotopes), or (2) the total uranium action level of one milligram of total uranium inhaled in one week.
- Each Shift, following modification of 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) 1 DAC (most likely uranium isotopes), or (2) the total uranium action level of one milligram inhaled by a worker in one week.

The representativeness of the air samplers shall be checked annually and when significant process or equipment changes have been made. Facility procedures specify how representativeness is determined.

Plant areas surveyed as described in this section include as a minimum UF_6 processing areas, decontamination areas, waste processing areas and laboratories. Continuous air monitors (e.g., stationary samplers or personnel lapel samplers) may be substituted when appropriate, as when continuous monitoring may not be reasonably achieved.

Action levels are based on trending of data collected during facility operation. Investigations are performed if airborne activity:

A. Exceeds 1 DAC (most likely uranium isotopes).

B. Shows a short-term increase of a factor of 10 over historical data from the previous 12 months.

Corrective actions include investigation of the adverse trend and an evaluation of the need for changes, consistent with the principles of ALARA.

4.8.2 External Exposures

The potential for significant external exposure to personnel under routine operating conditions is less significant than that for internal exposures. This is primarily due to the nature of the radionuclides present in the facility.

Parameters important in determining dose from external exposures are:

- The length of time the worker remains in the radiation field
- The intensity of the radiation field
- The portion of the body receiving the dose.

Historical data from European facilities of similar construction and representative operations show relatively low doses compared to nuclear power plant doses.

4.8.3 Procedures

Procedures are provided in the following areas to administratively control personnel radiation exposure:

- Operation
- Design
- Maintenance
- Modification
- Decontamination
- Surveillance

- Procurement.

4.8.4 Instrumentation

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

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 part of nuclear safety worker training.

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 part of nuclear safety worker training. Hand and foot monitors are used in applications where "pass-throughs" are frequent and where hand and foot monitoring is the major requirement.

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. A contamination monitor is placed at the local control point within the RCA. Personnel exiting the RCA are required to use a full body contamination monitor to check for contamination on their body. If the PCM is out of service an alternative method of monitoring is required (e.g. friskers).

4.8.5 Contamination Control

Small contamination areas may be roped off or otherwise segregated from the rest of an RCA. Appropriate clothing and/or other equipment is used to minimize exposure to radioactive material and prevent the spread of contamination. The entire RCA is not posted as a Contaminated Area.

4.8.5.1 Surface Contamination

Contamination surveys are performed in all UF_6 process areas. Additional routine surveys are performed in non-UF_6 process areas, including selected areas normally not suspected to be contaminated. Monitoring includes direct radiation and removable contamination measurements. Survey procedures are based on the potential for contamination of an area and operational experience. Selected areas within RCA are surveyed at least weekly. The lunch room and change rooms are outside the RCA and are surveyed at least weekly.

Removable surface contamination present on a surface 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 12.2.3.4) The Decontamination Workshop and Decontamination System are located in the same room in the CRDB. 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 (UF_6), uranium tetrafluoride (UF_4) and uranyl fluoride (UO_2F_2).

One of the functions of the Decontamination Workshop is to provide a maintenance facility for both UF_6 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.

4.9.2 Contaminated Material Handling Room

The Contaminated Material Handling Room, located in the CRDB, provides an area for storage of 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.

4.10 Decontamination Policy and Provisions

Removing radioactive material from equipment, to the extent reasonably possible prior to servicing reduces exposures to personnel who work around and service contaminated equipment. Surface contamination is removed to minimize its spread to other areas of the facility. Surfaces such as floors and walls are designed to be smooth, nonporous and free of cracks so that they can be more easily decontaminated.

Decontamination facilities and procedures for the CRDB and the SBMs have been discussed above. For the remaining areas of the SBMs, CRDB, and CAB, decontamination requirements involve only localized clean-up at areas where maintenance has been or is being performed that involves opening a uranium-containing system.

The facility follows 10 CFR 20, Subpart E, for the abandonment or release for unrestricted use of surfaces and premises.

4.11 Additional Program Commitments

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

4.11.1 Leak-Testing Byproduct Material Sources

In addition to the uranium processed at the facility, other sources of radioactivity are used. These sources are small calibration sources used for instrument calibration and response checking. These byproduct material sources may be in solid, liquid, or gaseous form; the sources may be sealed or unsealed. Both types of sources present a small radiation exposure risk to facility workers. For limits of possession for radioactive material types, quantities, and forms see current version of SNM-2010. Leak-testing of sources, available for use, is performed semi-annually using standard wipe protocols. Sources found to be leaking (contamination levels ≥0.005µCi) shall be removed from service and properly disposed of.

4.11.2 Records and Reports

The facility meets the following regulations for the additional program commitments applicable to records and reports:

- 10 CFR 20 (CFR, 2003b), Subpart L (Records), Subpart M (Reports)
- Section 70.61 (Performance requirements) (CFR, 2003e)
- Section 70.74 (Additional reporting requirements) (CFR, 2003s).

The facility Records Management program is described in Section 11.7, Records Management. The facility maintains records of the radiation protection program (including program provisions, audits, and reviews of the program content and implementation), radiation survey results (air sampling, bioassays, external-exposure data from monitoring of individuals, internal intakes of radioactive material), and results of corrective action program referrals, RWPs and planned special exposures. The facility maintains complete records of the Radiation Protection Program for at least the life of the facility.

By procedure, the facility will report to the NRC, within the time specified in 10 CFR 20.2202 (CFR, 2003t) and 10 CFR 70.74 (CFR, 2003s), any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR 20 (CFR, 2003b). The facility will prepare and submit to the NRC an annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b) (CFR, 2003u).

As previously noted in this chapter, LES will refer to the facility's corrective action program any radiation incident that results in an occupational exposure that exceeds the dose limits in 10 CFR 20, Appendix B (CFR, 2003m), or is required to be reported per 10 CFR 70.74 (CFR, 2003s). The facility reports to the NRC both the corrective action taken (or planned) to protect against a recurrence and the proposed schedule to achieve compliance with the applicable license condition or conditions.

4.12 References

Edition of Codes, Standards, NRC Documents, etc that are not listed below are given in ISAS Table 3.0-1.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 19, Notices, Instructions, and Reports to Workers: Inspections and Investigations, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Part 20, Standards for Protection Against Radiation, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 20.1101, Radiation protection programs, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Section 20.1201, Occupational dose limits for adults, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Section 20.1208, Dose equivalent to an embryo/fetus, 2003.

CFR, 2003h. Title 10, Code of Federal Regulations, Section 70.22, Contents of applications, 2003.

CFR, 2003i. Title 10, Code of Federal Regulations, Section 19.12, Instructions to workers, 2003.

CFR, 2003j. Title 10, Code of Federal Regulations, Section 20.2110, Form of records, 2003.

CFR, 2003k. Title 10, Code of Federal Regulations, Section 19.13, Notifications and reports to individuals, 2003.

CFR, 2003l. Title 29, Code of Federal Regulations, Part 1910, Occupational Safety and Health Standards, 2003.

CFR, 2003m. Title 10, Code of Federal Regulations, Part 20, Appendix B, Annual Limits on Intakes (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage, 2003.

CFR, 2003n. Title 10, Code of Federal Regulations, Section 20.1003, Definitions, 2003.

CFR, 2003o. Title 10, Code of Federal Regulations, Section 20.1301, Dose limits for individual members of the public, 2003.

4.12 References

CFR, 2003p. Title 40, Code of Federal Regulations, Part 190, Environmental Radiation Protection Standard For Nuclear Power Operations, 2003.

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.

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

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) Excludes accident situations
b) No routine extremity or skin monitoring is required
c) TEDE is the sum of internal dose and external dose received during routine operations
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 Modules)	< 1 E-4 (< 0.01)
Separations Building Module 1001 – Cascade Halls	5 E-4 (0.05)
Separations Building Module 1001 – UF_6 Handling Area & Process Services Corridor	1 E-3 (0.1)
Separations Building Module 1003 – Cascade Halls	TBD
Separations Building Module 1003 - UF_6 Handling Area & Process Services Corridor	TBD
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 4.7-1 Radiation Emitted from Natural UF_6 Feed

Element	Nuclide Symbol	Half-Life	Maximum Radiation Energies (Mev) and intensities		
			alpha (α)	beta (β)	gamma (γ)
92 uranium	^{238}U	4.5E+9 yr	4.15 25%	none	0.013 8.8%
90 thorium	^{231}Th	26 hr	4.20 75%	0.39 ~100%	0.025 14.7%
90 thorium	^{234}Th	24 d	none	0.19 73% 0.10 27%	0.06 3.8% 0.09 5.4%
91 protactinium	^{234}Pa	1.2 min	none	2.28 99%	0.766 0.21% 1.001 0.60%
92 uranium	^{234}U	2.5E+5 yr	4.72 28% 4.78 72%	none	0.053 0.12%
92 uranium	^{235}U	7.04E+8 yr	4.37 17% 4.40 55% 4.60 14%	none	0.143 12% 0.185 54% 0.205 6%

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

4.14 Chapter 4 Figures

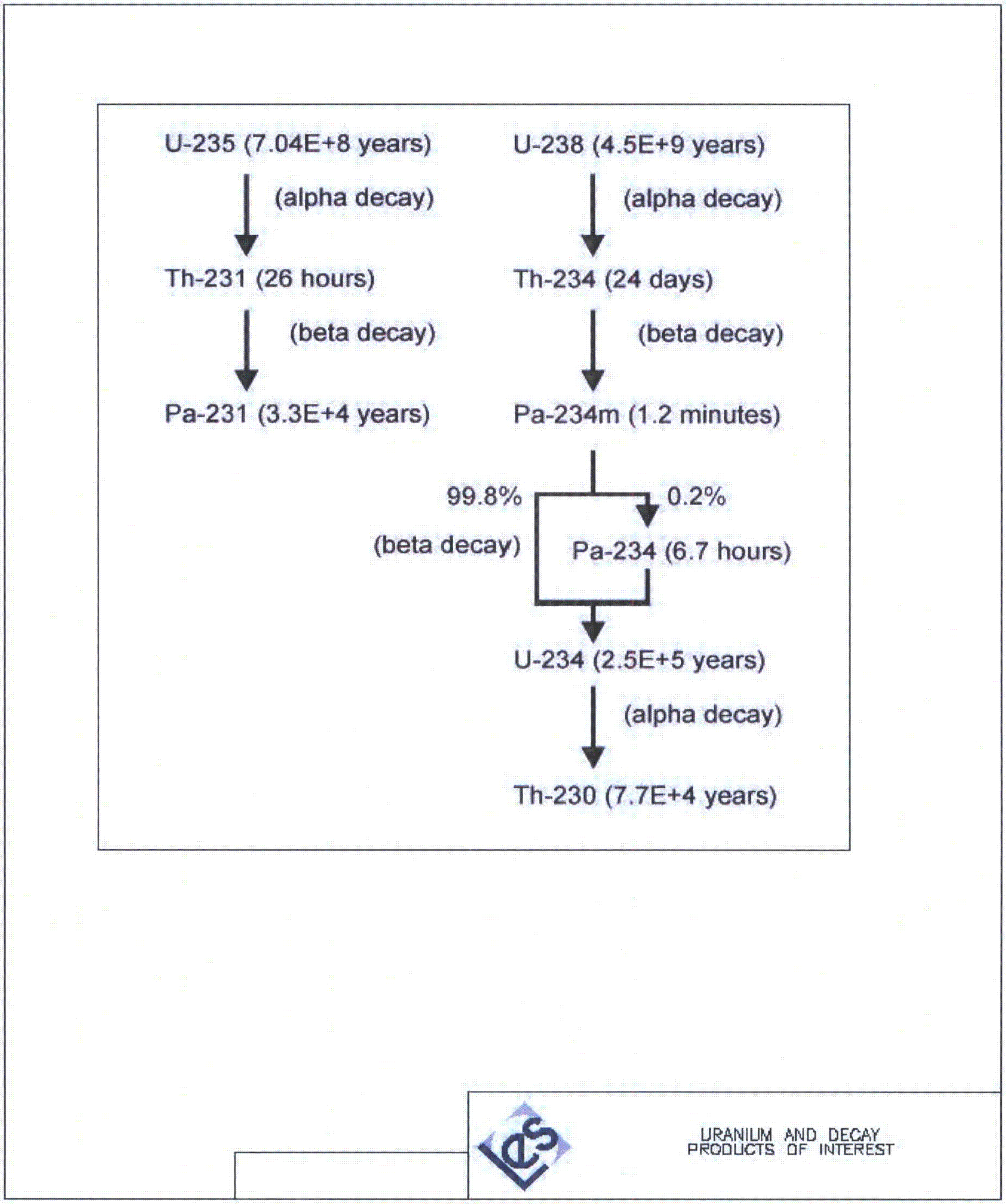


Figure 4.7-1 Uranium and Decay Products of Interest

4.14 Chapter 4 Figures

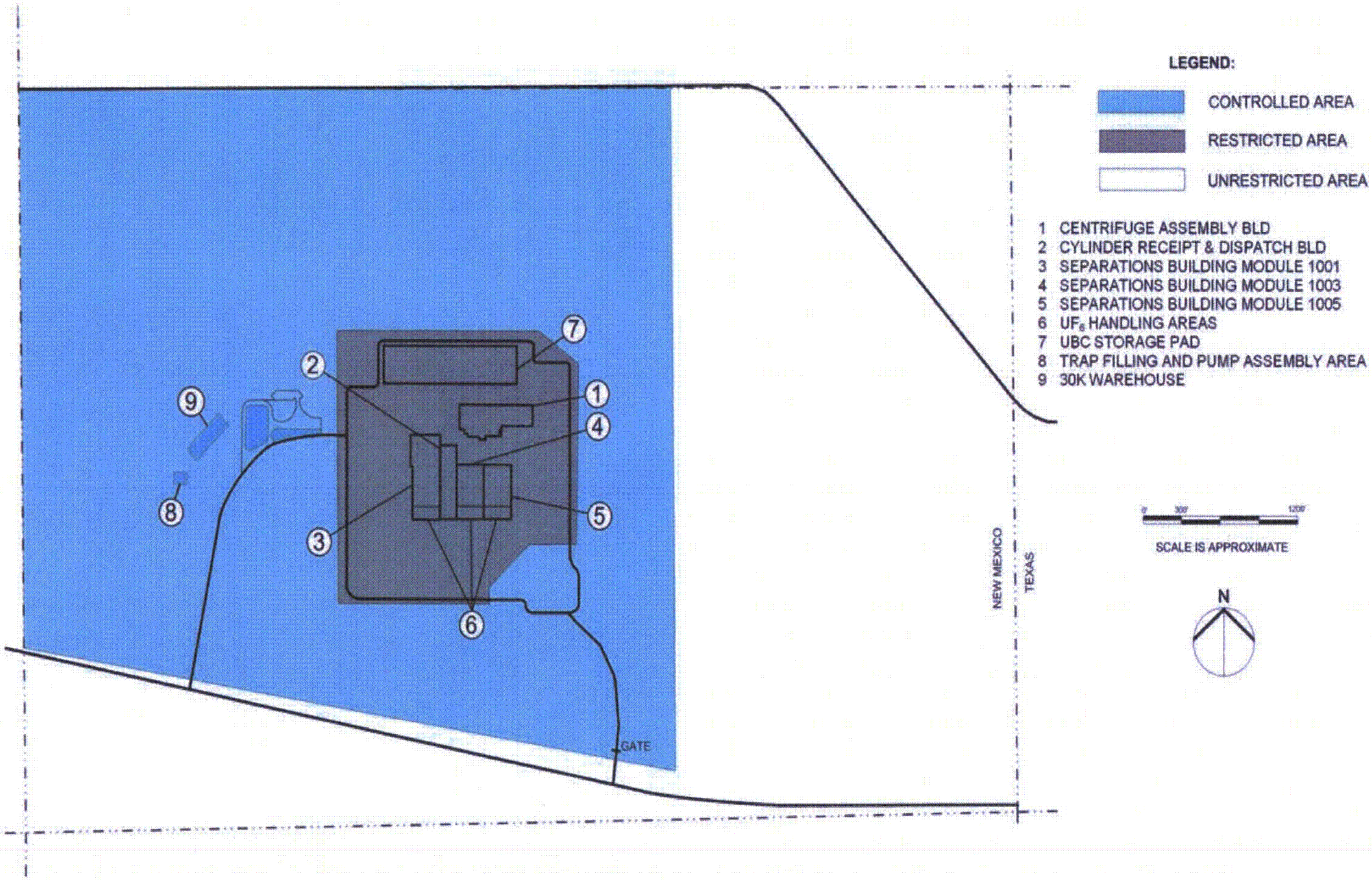


Figure 4.7-2 Projected Radiological Zones

5.0 Nuclear Criticality Safety

The Nuclear Criticality Safety Program for the National Enrichment Facility (NEF) is in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities. Regulatory Guide 3.71 provides guidance on complying with the applicable portions of NRC regulations, including 10 CFR 70 (CFR, 2003a), by describing procedures for preventing nuclear criticality accidents in operations involving handling, processing, storing, and transporting special nuclear material (SNM) at fuel and material facilities. The facility is committed to following the guidelines in this regulatory guide for specific ANSI/ANS criticality safety standards with the exception of ANSI/ANS-8.9, "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Material." Piping configurations containing aqueous solutions of fissile material will be evaluated in accordance with ANSI/ANS-8.1, using validated methods to determine subcritical limits.

The information provided in this chapter, the corresponding regulatory requirements, and the section of NUREG-1520, Chapter 5 in which the NRC acceptance criteria are presented is summarized below.

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 5 Reference
Section 5.1 Nuclear Criticality Safety (NCS) Program		
Management of the NCS Program	70.61(d) 70.64(a)	5.4.3.1
Control Methods for Prevention of Criticality	70.61	5.4.3.4.2
Safe Margins Against Criticality	70.61	5.4.3.4.2
Description of Safety Criteria	70.61	5.4.3.4.2
Organization and Administration	70.61	5.4.3.2
Section 5.2 Methodologies and Technical Practices		
Methodology	70.61	5.4.3.4.1 5.4.3.4.4 5.4.3.4.6
Section 5.3 Criticality Accident Alarm System (CAAS)		
Criticality Accident Alarm System	70.24	5.4.3.4.3
Section 5.4 Reporting		
Reporting Requirements	Appendix A	5.4.3.4.7 (7)

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 uranium enriched in isotope ^{235}U no greater than the LES license limit. However, as additional conservatism, most nuclear criticality safety analyses for enriched material are performed assuming a ^{235}U enrichment of 6.0 $^{w}/_{o}$, 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 $^{w}/_{o}$. 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 UF_6 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 UF_6 piping. UF_6 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.

- The NCS methodologies and technical practices will be kept applicable to current configuration by means of the configuration management function. The NCS program will be upgraded, as necessary, to reflect changes in the ISA or NCS methodologies and to modify operating and maintenance procedures in ways that could reduce the likelihood of occurrence of an inadvertent nuclear criticality.
- The NCS program will be used to establish and maintain NCS safety limits and NCS operating limits for IROFS in nuclear processes and a commitment to maintain adequate management measures to ensure the availability and reliability of the IROFS.
- NCS postings will be provided and maintained current.
- NCS emergency procedure training will be provided.
- The NCS baseline design criteria requirements in 10 CFR 70.64(a) (CFR, 2003c) will be adhered to.
- The NCS program will be used to evaluate modifications to operations, to recommend process parameter changes necessary to maintain the safe operation of the facility, and to select appropriate IROFS and management measures.
- The NCS program will be used to promptly detect NCS deficiencies by means of operational inspections, audits, and investigations. Deficiencies will be entered into the corrective action program so as to prevent recurrence of unacceptable performance deficiencies in IROFS, NCS function or management measures.
- NCS program records will be retained as described in Section 11.7, Records Management.

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

- Appreciation of the physics of nuclear criticality safety.
- Information obtained from the analysis of jobs and tasks in accordance with Section 11.3.

Additional discussion of management measures is provided in Chapter 11, Management Measures.

5.1.2 Control Methods for Prevention of Criticality

The major controlling parameters used in the facility are enrichment control, geometry control, moderation control, and/or limitations on the mass as a function of enrichment. In addition, reflection, interaction, and heterogeneous effects are important parameters considered and applied where appropriate in nuclear criticality safety analyses. Nuclear Criticality Safety Evaluations and Analyses are used to identify the significant parameters affected within a particular system. All assumptions relating to process, equipment, material function, and operation, including credible abnormal conditions, are justified, documented, and independently reviewed. Where possible, passive engineered controls are used to ensure nuclear criticality safety. The determination of the safe values of the major controlling parameters used to control criticality in the facility is described below.

Moderation control is in accordance with ANSI/ANS-8.22, Nuclear Criticality Safety Based on Limiting and Controlling Moderators. However, for the purposes of the criticality analyses, it is 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 MONK 8A 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 $^w/_o$ and 6.0 $^w/_o$.

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 $^w/_o$ enrichment, as additional conservatism, the values in Table 5.1-2, Safety Criteria for Buildings/Systems/ Components, represent the limits based on 6.0 $^w/_o$ enrichment except for the Contingency Dump System equipment and piping on the 2nd floor of the Process Services Area and the Tails Take-off System which are limited to 1.5 $^w/_o$ ^{235}U.

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 to a maximum of the LES license limit except for the systems and components associated with a cascade dump. For added conservatism the systems controlled to the LES license limit in isotope ^{235}U are analyzed at 6%.
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%

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_{eff} = kcalc + 3 σcalc < 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 $^{w}/_{o}$ ^{235}U for safety margin.

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.

Reflection

Reflection is considered when performing Nuclear Criticality Safety Evaluations and Analyses. The possibility of full water reflection is considered but the layout of the NEF is a very open design and it is highly unlikely that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. In addition, automatic sprinklers are excluded from SBMs and the CRDB. Fire protection standpipes are located in enclosed stairwells, or are arranged such that flooding from these sources is highly unlikely. Therefore, full water reflection of vessels has therefore been discounted. However, some select analyses have been performed using full reflection for conservatism. Partial reflection of

2.5 cm (0.984 in) of water is assumed where limited moderating materials (including humans) may be present It is recognized that concrete can be a more efficient reflector than water; therefore, it is modeled in analyses where it is present. When moderation control is identified in the ISA Summary, it is established consistent with the guidelines of ANSI/ANS-8.22.

Interaction

Nuclear criticality safety evaluations and analyses consider the potential effects of interaction. A non-interacting unit is defined as a unit that is spaced an approved distance from other units such that the multiplication of the subject unit is not increased. Units may be considered non-interacting when they are separated by more than 60 cm (23.6 inches).

If a unit is considered interacting, nuclear criticality safety analyses are performed. Individual unit multiplication and array interaction are evaluated using the Monte Carlo computer code MONK 8A to ensure k_{eff} = kcalc + 3 σcalc < 0.95.

Neutron Absorbers

Neutron Absorption is a factor in almost all of the materials at the NEF. The normal absorption of neutrons in standard materials used in the construction and processes at the NEF (uranium, fluorine, water, steel, etc.) is not specifically excluded as a criticality control parameter.

Models incorporate conservative values (e.g., material compositions and equipment dimensions), which are validated at receipt, after installation or during surveillances.

Additional materials such as cadmium and boron for which the sole purpose would be to absorb neutrons are not incorporated in NEF processes. Solutions of absorbers are not used as a criticality control mechanism.

Concentration and Density

NEF does not use either concentration or density as a criticality control parameter.

5.1.3 Safe Margins against Criticality

Process operations require establishment of criticality safety limits. The facility UF_6 systems involve mostly gaseous operations. These operations are carried out under reduced atmospheric conditions (vacuum) or at slightly elevated pressures not exceeding three 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 Modules, significant accumulations of enriched UF_6 reside only in the Product Low Temperature Take-off Stations, Product Liquid Sampling Autoclaves, Product Blending System or the UF_6 cold traps. All these, except the UF_6 cold traps (which are safe-by-design), contain the UF_6 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 UF_6. In addition, the facility's stringent procedural controls for enriching UF_6 assure that it does not become unacceptably hydrogen moderated while in process. The plant's UF_6 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 UO_2F_2, are applied to the facility to prevent a nuclear criticality event. Although the NEF will be limited to Material License Condition 6B for $^{w}/_{o}$ 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%.

Where there are significant in-process accumulations of enriched uranium as UF_6, 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, Safety, and Environmental Manager as described in Chapter 2, Organization and Administration. The Health, Safety, and Environmental 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
- 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.

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 Criticality Safety Engineer training program is based on ANSI/ANS-8.26, Criticality Safety Engineer Training and Qualification Program. 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

This section describes the methodologies and technical practices used to perform the Nuclear Criticality Safety (NCS) analyses and NCS evaluations. The determination of the NCS controlled parameters and their application and the determination of the NCS limits on IROFS are also presented.

5.2.1 Methodology

MONK 8A is a powerful Monte Carlo tool for nuclear criticality safety analysis. The advanced geometry modeling capability and detailed continuous energy collision modeling treatments provide realistic 3-dimensional models for an accurate simulation of neutronic behavior to provide the best estimate neutron multiplication factor, k-effective. Complex models can be simply set up and verified. Additionally, MONK 8A has demonstrable accuracy over a wide range of applications and is distributed with a validation database comprising critical experiments covering uranium, plutonium and mixed systems over a wide range of moderation and reflection. The experiments selected are regarded as being representative of systems that are widely encountered in the nuclear industry, particularly with respect to chemical plant operations, transportation and storage. The validation database is subject to on-going review and enhancement. A categorization option is available in MONK 8A to assist the criticality analyst in determining the type of system being assessed and provides a quick check that a calculation is adequately covered by validation cases.

5.2.1.1 Methods Validation

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

The extensive validation database contains a number of experiments applicable to this application involving low and intermediate-enriched uranium. The MONK 8A code with the JEF2.2 library was validated against these experiments which are provided in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (NEA, 2002) and NUREG/CR-1071. The experiments chosen are provided in Table 5.2-1, Uranium Experiments Used for Validation, along with a brief description. The overall mean calculated value from these 93 configurations is 1.0017 ± 0.0045 and the results are provided in the MONK 8A Validation and Verification report.

MONK 8A is distributed in ready-to-run executable form. This approach provides the user with a level of quality assurance consistent with the needs of safety analysis. The traceability from source code to executable code is maintained by the code vendor.

In accordance with the guidance in NUREG-1520, code validation for the specific application has been performed. Specifically, the experiments provided in Table 5.2-1, Uranium Experiments Used for Validation, were calculated and documented in the MONK 8A Validation and Verification report for the National Enrichment Facility. In addition, the MONK 8A Validation and Verification report 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. Any revision to the validation of neutron transport methods will be performed using ANSI/ANS-8.24, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations as a guideline with exception as identified in Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities (Revision 2.2010). The two exceptions pertain to the use of a positive bias and rejection of outliers.

The MONK 8A 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:

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

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 σBias from the MONK 8A Validation and Verification is 0.0085 and a value of 0.05 is assigned to the subcritical margin, ΔSM. The term ΔAOA 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 ΔAOA 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 ΔAOA 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:

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

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

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

Enrichment is controlled to limit the percent ^{235}U within any process vessel or container to the LES license limit except for the systems and components associated with a cascade dump. For added conservatism the systems controlled to the LES license limit in isotope ^{235}U 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%

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 UF_6 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 UF_6 and water vapor in the presence of excess UF_6 can be represented by the equation:

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

Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound. 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 $UO_2F_2 \cdot 1.5H_2O$ is formed and, additionally, that the HF produced by the UF_6/water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:

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

For the MONK 8A calculations, the composition of the breakdown product was simplified to $UO_2F_2 \cdot 3.5H_2O$ that gives the same H/U ratio of 7 as above.

In the case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated perfluorinated polyether (PFPE) type lubricant. Mixtures of UF_6 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 $^{w}/_{o}$. Therefore, the uranyl fluoride/water mixture assumption provides additional conservatism in this case.

5.2.1.3.4 Vessel Movement Assumption

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.

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.

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 MONK 8A. This adequately covers all models likely to be purchased.
- The UF_6 pumping units are a combination unit of two pumps, one 500 m^3/hr (17,656 ft^3/hr) pump with a free volume of 8.52 L (2.25 gal) modeled as a cylinder, and a larger 2000 m^3/hr (70,626 ft^3/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 UO_2F_2, 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 UF_6 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.
- A discussion of the analysis results, including identification of required limits and controls.

During the design, construction and operations phases of NEF, the NCS analysis is performed by a criticality safety engineer and independently reviewed by a second criticality safety engineer. During the operation of NEF, the NCS analysis is performed by a criticality safety engineer, independently reviewed by a second criticality safety engineer and approved by the Engineering Manger or the Criticality Safety Engineering Supervisor. Only qualified criticality safety engineers can perform NCS analyses and associated independent review.

5.2.1.5 Additional Nuclear Criticality Safety Analyses Commitments

The NEF NCS analyses were performed using the above methodologies and assumptions. NCS analyses also meet the following:

- NCS analyses are performed using acceptable methodologies.
- Methods are validated and used only within demonstrated acceptable ranges.
- The analyses adhere to ANSI/ANS-8.1 as it relates to methodologies.
- The validation report statement in Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities, is as follows: LES has demonstrated (1) the adequacy of the margin of safety for subcriticality by assuring that the margin is large compared to the uncertainty in the calculated value of k_{eff}, (2) that the calculation of k_{eff} is based on a set of variables whose values lie in a range for which the methodology used to determine k_{eff} has been validated, and (3) that trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.
- A specific reference to (including the date and revision number) and summary description of either a manual or a documented, reviewed, and approved validation report for each methodology are included. Any change in the reference manual or validation report will be reported to the NRC by letter.
- The reference manual and documented reviewed validation report will be kept at the facility.
- The reference manual and validation report are incorporated into the configuration management program.
- The NCS analyses are performed in accordance with the methods specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520, Section 5.4.3.4, are used to analyze NCS accident sequences in operations and processes.
- The acceptance criteria in NUREG-1520, Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.
- As stated in ANSI/ANS-8.1, process specifications incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.
- ANSI/ANS-8.7, as it relates to the requirements for subcriticality of operations, the margin of subcriticality for safety, and the selection of controls required by 10 CFR 70.61(d) (CFR, 2003b), is used.

- ANSI/ANS-8.10, as modified by Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuel and Material Facilities, as it relates to the determination of consequences of NCS accident sequences, is used.
- If administrative k_{eff} margins for normal and credible abnormal conditions are used, NRC pre-approval of the administrative margins will be sought.
- Subcritical limits for k_{eff} calculations such that: k_{eff} subcritical = 1.0 - bias - margin, where the margin includes adequate allowance for uncertainty in the methodology, data, and bias to assure subcriticality are used.
- Studies to correlate the change in a value of a controlled parameter and its k_{eff} value are performed. The studies include changing the value of one controlled parameter and determining its effect on another controlled parameter and k_{eff}.
- The double contingency principle is met. The double contingency principle is used in determining NCS controls and IROFS.
- The acceptance criteria in NUREG-1520 Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, are met.

5.2.1.6 Nuclear Criticality Safety Evaluations (NCSE)

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures), that involves or could affect uranium, a NCSE shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with approved margin for safety) under both normal and credible abnormal conditions. If this condition cannot be shown with the NCSE, either a new or revised NCS analysis will be generated that meets the criteria, or the change will not be made.

The NCSE shall determine and explicitly identify the controlled parameters and associated limits upon which NCS depends, assuring that no single inadvertent departure from a procedure could cause an inadvertent nuclear criticality and that the safety basis of the facility will be maintained during the lifetime of the facility. The evaluation ensures that all potentially affected uranic processes are evaluated to determine the effect of the change on the safety basis of the process, including the effect on bounding process assumptions, on the reliability and availability of NCS controls, and on the NCS of connected processes.

The NCSE process involves a review of the proposed change, discussions with the subject matter experts to determine the processes which need to be considered, development of the controls necessary to meet the double contingency principle, and identification of the assumptions and equipment (e.g., physical controls and/or management measures) needed to ensure criticality safety.

Engineering judgment of the criticality safety engineer is used to ascertain the criticality impact of the proposed change. The basis for this judgment is documented with sufficient detail in the NCSE to allow the independent review by a second criticality safety engineer to confirm the conclusions of the judgment of results. Each NCSE includes, as a minimum, the following information.

- A discussion of the scope of the evaluation, a description of the system(s)/process(es) being evaluated, and identification of the applicable nuclear criticality safety analysis

- A discussion to demonstrate the applicable nuclear criticality safety analysis is bounding for the condition evaluated.
- A discussion of the impact on the facility criticality safety basis, including effect on bounding process assumptions, on reliability and availability NCS controls, and on the nuclear criticality safety of connected system(s)/process(es).
- A discussion of the evaluation results, including (1) identification of assumptions and equipment needed to ensure nuclear criticality safety is maintained and (2) identification of limits and controls necessary to ensure the double contingency principle is maintained.

The NCSE is performed and documented by a criticality safety engineer. Once the NCSE is completed and the independent review by a criticality safety engineer is performed and documented, the Engineering Manager or the Criticality Safety Engineering Supervisor approves the NCSE. Only criticality safety engineers who have successfully met the requirements specified in the qualification procedure can perform NCSEs and associated independent review.

The above process for NCSEs is in accordance with ANSI/ANS 8.19.

5.2.1.7 Additional Nuclear Criticality Safety Evaluations Commitments

NCSEs also meet the following:

- The NCSEs are performed in accordance with the procedures specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520, Sections 5.4.3.4.1(10)(a), (b), (d) and (e), are used to evaluate NCS accident sequences in operations and processes.
- The acceptance criteria in NUREG-1520, Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.
- The double contingency principle is met. The double contingency principle is used in determining NCS controls and IROFS.
- The acceptance criteria in NUREG-1520 Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, are met.

5.3 Criticality Accident Alarm System (CAAS)

The facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, (CFR, 2003d). Areas where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2003d) mass limits are provided with CAAS coverage. Emergency management measures are covered in the facility Emergency Plan.

5.4 Reporting

The following are NCS Program commitments related to event reporting:

- A program for evaluating the criticality significance of NCS events will be provided and an apparatus will be in place for making the required notification to the NRC Operations Center. Qualified individuals will make the determination of significance of NCS events. The determination of loss or degradation of IROFS or double contingency principle compliance will be made against the license and 10 CFR 70 Appendix A (CFR, 2003f).
- The reporting criteria of 10 CFR 70 Appendix A and the report content requirements of 10 CFR 70.50 (CFR, 2003g) will be incorporated into the facility emergency procedures.
- The necessary report based on whether the IROFS credited were lost, irrespective of whether the safety limits of the associated parameters were actually exceeded will be issued.
- If it cannot be ascertained within one hour of whether the criteria of 10 CFR 70 Appendix A (CFR, 2003f) Paragraph (a) or (b) apply, the event will be treated as a one-hour reportable event.

5.5 References

Edition of Codes, Standards, NRC Documents, etc., that are not listed below are given in ISAS Table 3.0-1.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.24, Criticality accident requirements, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.72, Facility changes and change process, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Part 70, Appendix A, Reportable Safety Events, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Section 70.50, Reporting requirements, 2003.

Lychev, 1990. Crystalline Hydrates of Uranyl Fluoride at 20°C, Lychev, Mikhalev and Suglobov. Journal of Soviet Radiochemistry, Vol 32, 1990.

NEA, 2002. International Handbook of Evaluated Criticality Safety Benchmark Experiments, NEA/NSC/DOC(95)03, Nuclear Energy Agency, September 2002 Edition.

SA, 2001. Serco Assurance, ANSWERS Software Service, "Users Guide for Version 8 ANSWERS/MONK(98) 6," 1987-2001.

5.6 Chapter 5 Table

5.6 Chapter 5 Table

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 $^{w}/_{o}$ enrichment			
Volume	30.3 L (8.0 gal)	22.9 L (6.1 gal)	0.76
Cylinder Diameter	26.6cm(10.5 in)	23.9 cm (9.4 in)	0.90
Slab Thickness	12.8 cm (5.0 in)	11.1 cm (4.4 in)	0.87
Water Mass	18.5 kg H_2O (40.8 lb H_2O)	14.2 kg H_2O (31.1 lb H_2O)	0.77
Areal Density	11.8 g/cm^2 (24.2 lb/ft^2)	9.9g/cm^2 (20.3 lb/ft^2)	0.84
Uranium Mass	36.7 kg U (80.9 lb U)		
- no double batching		26.8 kg U (59.1 lb U)	0.73
- double batching		16.5 kg U (36.4 lb U)	0.45
Values for 6.0 $^{w}/_{o}$ enrichment			
Volume	25.3 L (6.7 gal)	19.3 L (5.1 gal)	0.76
Cylinder Diameter	24.8 cm (9.8 in)	22.4 cm (8.8 in)	0.90
Slab Thickness	11.6 cm (4.6in)	10.1 cm (4.0 in)	0.87
Water Mass	15.4 kg H_2O (34.0 lb H_2O)	11.9 kg H_2O (26.2 lb H_2O)	0.77
Areal Density	9.4 g/cm^2 (19.3 lb/ft^2)	7.9 g/cm^2 (16.2 lb/ft^2)	0.84
Uranium Mass	27 kg U (59.5 lb U)		
- no double batching		20.1 kg U (29.7 kg UF_6)	0.74
- double batching		12.2 kg U (26.9 lb U)	0.45

5.6 Chapter 5 Table

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	< 22.4 cm (8.8 in)
Product Cylinders (30B)	Moderation	H < 0.98 kg (2.16 lb)
UF_6 Piping	Diameter	< 22.4 cm (8.8 in)
Chemical Traps	Diameter	< 22.4 cm (8.8 in)
Product Cold Trap	Diameter	< 22.4 cm (8.8 in)
Contingency Dump System 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	Volume	< 19.3 L (5.1 gal)
Individual Uranic Liquid Containers, e.g., PFPE Oil Bottle, Laboratory Flask, Mop Bucket	Volume	< 19.3 L (5.1 gal)
Vacuum Cleaners Oil Containers	Volume	<19.3 L (5.1 gal)

5.6 Chapter 5 Table

Table 5.2-1 Uranium Experiments Used for Validation

MONK8A Case	Case Description	Number of Experiments	Handbook Reference
25	Low-enriched damp U_3O_8 powder in cubic aluminum cans	10	NUREG/CR-1071
42	MARACAS Program: Polythene reflected critical configurations with low enriched and low moderated uranium dioxide powder U(5) O_2	18	LEU-COMP-THERM-049
43	Low-enriched uranyl nitrate solutions	3	LEU-SOL-THERM-002
51	Low-enriched uranium solutions (new STACY experiments)	7	LEU-SOL-THERM-004
63	Boron carbide absorber rods in uranyl nitrate (5.6 $^w/_o$ enriched)	3	LEU-SOL-THERM-005
69	Critical arrays of polyethylene-moderated U(30)F_4-Polytetrafluoroethylene one-inch cubes	29	IEU-COMP-THERM-001
71	STACY: 28 cm thick slabs of 10 $^w/_o$ enriched uranyl nitrate solutions, water reflected	7	LEU-SOL-THERM-016
80	STACY: Unreflected 10 $^w/_o$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	5	LEU-SOL-THERM-007
81	STACY: Concrete reflected 10 $^w/_o$ enriched uranyl nitrate solution reflected by concrete	4	LEU-SOL-THERM-008
84	STACY: Borated concrete reflected 10 $^w/_o$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	3	LEU-SOL-THERM-009
85	STACY: Polyethylene reflected 10 $^w/_o$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	4	LEU-SOL-THERM-010

6.0 Chemical Process Safety

This chapter describes the Louisiana Energy Services (LES) plan for managing chemical process safety and demonstrating that chemical process safety controls meet the requirements of 10 CFR 70 (CFR, 2003a) thereby providing reasonable assurance that the health and safety of the public and facility employees is protected. The chapter describes the chemical classification process, the hazards of chemicals of concern, process interactions with chemicals affecting licensed material and/or hazardous chemicals produced from licensed material, the methodology for evaluating hazardous chemical consequences, and the chemical safety assurance features.

The NEF chemical process safety program meets the acceptance criteria in Chapter 6 of NUREG-1520 and complies with 10 CFR 70.61 (CFR, 2003b), 70.62 (CFR, 2003c) and 70.64 (CFR, 2003d).

The information provided in this chapter, the corresponding regulatory requirement and the section of NUREG-1520 Chapter 6 in which the NRC acceptance criteria are presented are summarized below:

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 5 Reference
Section 6.1 Chemical Information		
• Properties and Hazards	70.62(c)(1)(ii)	6.4.3.1
Section 6.2 Chemical Process Information		
• General Information	70.65(b)(3)	6.4.3.1
• Design Basis, Materials, Parameters	70.62(b)	6.4.3.1
• Process Chemistry, Chemical Interaction		6.4.3.2
Section 6.3 Chemical Hazards Analysis		
• Methodology, Scenarios, Evaluation	70.65(b)(3)	6.4.3.2
Section 6.4 Chemical Safety Assurance		
• Management, Configuration Control, Design, BDC, Maintenance, Training, Procedures, Audits, Emergency Planning, Incident Investigation	70.65(b)(4)	6.4.3.2 6.4.3.3